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POWER REACTOR TECHNOLOGY

A Quarterly Technical Progress Review

Prepared for DIVISION OF TECHNICAL INFORMATION, USAEC, by

W. H. ZINN and J. R. DIETRICH, GENERAL NUCLEAR ENGINEERING CORPORATION

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Winter 1964-1965

Progress on Specific Reactor Types:

A Review of Third Geneva Conference Papers

● VOLUME 8

● NUMBER 1

TECHNICAL PROGRESS REVIEWS

To meet the needs of industry for concise summaries of current atomic developments, the Atomic Energy Commission is publishing this series, Technical Progress Reviews. Issued quarterly, each of the reviews digests and evaluates the latest findings in a specific area of nuclear technology and science.

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Isotopes and Radiation Technology, P. S. Baker, A. F. Rupp, and associates, Oak Ridge National Laboratory

Nuclear Safety, Wm. B. Cottrell, W. H. Jordan, and associates, Oak Ridge National Laboratory

Power Reactor Technology, W. H. Zinn and J. R. Dietrich, Combustion Engineering, Inc., Nuclear Division

Reactor Fuel Processing, Stephen Lawroski and associates, Chemical Engineering Division, Argonne National Laboratory

Reactor Materials, R. W. Dayton, E. M. Simons, and associates, Battelle Memorial Institute

Each journal may be purchased from the Superintendent of Documents, U. S. Government Printing Office, Washington, D. C., 20462. *Isotopes and Radiation Technology* at \$2.00 per year for subscription or \$0.55 for individual issues; the other four journals at \$2.50 per year and \$0.70 per issue. See back cover for remittance instructions and foreign postage requirements.

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POWER REACTOR TECHNOLOGY

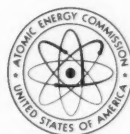
A REVIEW OF RECENT DEVELOPMENTS

Prepared for DIVISION OF TECHNICAL INFORMATION, USAEC,
by W. H. ZINN and J. R. DIETRICH,
COMBUSTION ENGINEERING, INC., NUCLEAR DIVISION

WINTER 1964-1965

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NUMBER 1



Foreword

This quarterly review of reactor development has been prepared at the request of the Division of Technical Information of the U. S. Atomic Energy Commission. Its purpose is to assist interested organizations in the task of keeping abreast of new results in reactor technology for civilian application.

Power Reactor Technology contains reviews of selected recently published reports that are judged noteworthy in the fields of power-reactor research and development, power-reactor applications, design practice, and operating experience. It is not meant to be a comprehensive abstract of all material published during the quarter, nor is it meant to be a treatise on any part of the subject. However, related reports from different sources are often treated together to yield reviews having some breadth of scope, and background material may be added to place recent developments in perspective. Occasionally the reviews are written by guest authors. Reviews having unusual breadth or significance are placed at the front of the issue as Feature Articles.

The intention is to cover the various areas of reactor development from the general viewpoint of the reactor designer rather than from the more detailed points of view of specialists in the individual areas. To whatever extent the coverage of *Power Reactor Technology* may occasionally overlap the fields of the other Technical Progress Reviews, the overlaps will be motivated by this objective of viewing current progress through the eyes of the reactor designer.

A degree of critical appraisal and some interpretation of results are often necessary to define the significance of reported work. Any such appraisal or interpretation represents only the opinion of the reviewer and (in the usual case, when the review is written by Combustion Engineering, Inc., Nuclear Division staff) the Editor. When the review is predominantly interpretive the reviewer is named; unless identified as a guest author, he is a member of the Combustion Engineering, Inc., Nuclear Division staff. Readers are urged to consult the original references to obtain all the background of the work reported and to obtain the interpretation of the results given by the original authors.

For timely coverage, *Power Reactor Technology* must often review fragmentary material. The fixed subject headings listed below have been adopted in the hope of maintaining some continuity and order in the material from one issue to another: all reviews except Feature Articles will be arranged under these headings. A particular issue will not necessarily contain all the headings but only those under which material is reviewed.

Economics, Applications, Programs
Resources and Fuel Cycles
Physics
Fluid and Thermal Technology
Fuel Elements
Materials
Control and Dynamics
Containment, Radiation Control, and Siting

Systems Technology
Components
Design and Construction Practice
Operating Experience
Specific Reactor Types
Specific Applications
Unconventional Approaches

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Availability of Third Geneva Conference Papers

1. Copies of all papers may be purchased from the European Office of the United Nations for two Swiss francs (46 cents) per paper plus postage. Address: European Office of the United Nations, Palais des Nations, Geneva, Switzerland, Attention: R. Furstenberg, Chief, Distribution Service Documents Division. When UN stocks are exhausted, reproduced copies of all papers may be purchased from the IAEA in Vienna at a price of \$1.50 per paper. Address: International Atomic Energy Agency, Kaerntnerring 11, Vienna I, Austria.

2. Reproduced copies of all U. S. papers (no foreign) may be purchased at a price of \$1.00 per paper from the Clearinghouse for Federal Scientific and Technical Information (formerly OTS). Address: National Bureau of Standards, U. S. Department of Commerce, 5285 Port Royal Road, Springfield, Va. 22151.

3. The American Nuclear Society will provide Xerox copies of all papers at 25 cents per page. Address: American Nuclear Society, 244 East Ogden Ave., Hinsdale, Ill. 60521.

4. The U. S. Atomic Energy Commission, Division of Technical Information Extension, Oak Ridge, Tenn. 37831, has produced micronegatives of all Geneva papers. Complete sets of duplicate negatives were distributed to the AEC, other government agencies, and their contractors; sets of U. S. papers only were distributed to AEC depository libraries.

(All U. S. papers were abstracted and indexed in Vol. 18, No. 18, of *Nuclear Science Abstracts*. All foreign papers will be similarly abstracted and indexed in Vol. 18, No. 21.)



Section

I

Power Reactor Technology

Introduction

Of the massive information collection produced by the Third United Nations International Conference on the Peaceful Uses of Atomic Energy, an important fraction deals with specific power-reactor types, with the progress made in their development, and with the experience gained in building and operating individual reactors of the various types. This entire issue of *Power Reactor Technology* is devoted to a review of that information.* More general reviews of the conference have been published elsewhere, notably in *Nucleonics*¹ and *Nuclear News*.²

Because of the special nature of this issue, the usual section headings as given in the Foreword do not appear. Instead, each of the following sections is on a particular reactor type.

Of the reactor types under serious development, two can be said to have achieved commercial acceptance: the "water" reactors (pressurized and boiling H₂O) and the "Mgnox" gas-cooled reactors. The heaviest building commitments have been made in the latter type, amounting, through 1968, to over 5000 Mw(e) in the United Kingdom and some 1300 Mw(e) in France. The application of the prestressed concrete pressure vessel to this reactor type has done much to improve its potentialities, with respect to both size and operating pressure. Nevertheless, it appears that other types will replace it in future building plans, at least in the United Kingdom. Whether the replacement type will be its direct descendant, the advanced gas-cooled reactor, is not clear at this time.

The water reactors are still in a state of vigorous development, and it appears that, in

the economic climate of the United States, it will be difficult for another reactor type to replace them so long as the basic costs of partially enriched nuclear fuels do not rise much above current levels. In the economic competition between the boiling- and pressurized-water types, it is difficult to detect a clear basic advantage of one over the other at the moment, although the rapid trend to plants of very high output may favor the pressurized-water type in the long run.

Another group of reactor types, each of which is represented by one or two bona fide power-generating plants, mostly in the lower capacity ranges, might be said to be in the "demonstration" stage. The most advanced of these are the Russian graphite-moderated superheat reactor (Beloyarsk) and the pressure-tube D₂O reactor [for each of which commitments have been made for a follow-on reactor in the 200-Mw(e) range] and the sodium-graphite reactor. The advanced gas-cooled reactor and the organic-moderated reactor also fall in this group. However, in regard to the latter, indications are that the future of organic liquids in central-station reactors lies in their use as coolants for D₂O-moderated reactors. The interesting variation on the organic-moderated concept represented by the Russian ARBUS reactor appears to be directed specifically to remote-station applications.

The remaining rather large group of reactor types under serious development cannot be said to be "demonstrated" in terms of prototypes of commercial versions. Some are very close to that stage, however. The integral nuclear superheat concept awaits only the full-scale operation of the BONUS and Pathfinder plants to enter the "demonstration" stage: both of these plants are currently in the process of startup. The heavy-water reactor of the pressure-vessel type is represented by two

*The Conference papers are to be published in bound form. Meanwhile, the papers are available and can be obtained from the sources listed on the preceding page.

operating plants: the Halden Boiling Water Reactor that produces process steam, and the Agesta plant, a pressurized heavy-water plant that produces 10 Mw of electricity along with hot water for space heating. It would be stretching a point, however, to call these demonstration plants in the sense of the word used here; for the proposed characteristics of the commercial version, as exemplified by the Marviken plant, involve substantial extrapolations in several areas. The high-temperature gas-cooled type has been critical (the Dragon reactor), and the Peach Bottom plant is under construction; the latter is intended as a prototype. At least three versions of D_2O -moderated pressure-tube reactors utilizing coolants other than D_2O are under serious development; the coolants used are H_2O (boiling liquid, fog, or steam), organic, and gas. Experimental reactors having the latter two coolants are being built.

The position of the fast-breeder reactor is a little hard to classify, but it can hardly be

said to have reached the demonstration stage despite the facts that it was first used to produce electric power in 1951, that a plant of moderately large output (Dounreay) has been in operation for some time, and that a large commercial plant (the EFFBR) is in process of startup. This is primarily because a fuel acceptable for commercial operation has not yet been demonstrated. Nevertheless, of all the reactor types, the fast breeder is probably the one of most universal interest, and continuing strong development efforts are foreseen for the future.

In the following sections the preceding rather brief remarks are expanded, and the status of each of the reactor types is reviewed on the basis of the information presented at Geneva.

References

1. *Nucleonics*, 22(10): (October 1964).
2. *Nucl. News*, 7(10): (October 1964).

Section

II

Power Reactor Technology

Boiling and Pressurized Light-Water Reactors

By John S. Wiley

The nonmilitary land-based boiling and pressurized light-water-moderated reactors of the world, other than the purely experimental ones, are listed in Table II-1. The table includes the more important reactors under construction as well as the reactors in operation. These reactor types are the ones of most interest, and the best known, in the United States; indeed, with the exception of the Russian reactor at Voronezh, all the reactors in Table II-1 were designed either by U. S. firms or with their assistance. Because these types have been covered most thoroughly by *Power Reactor Technology* in the past, this review of the many Geneva papers on light-water reactors is abbreviated in favor of greater coverage for the reactor types less well known in the United States. The major past reviews in *Power Reactor Technology*, of design and operating experience for specific water-moderated reactors, may be located by means of Table II-2.

Operating Experience

The operating experience with Yankee, the CETR, Saxton, and Shippingport is detailed in Ref. 1, and Ref. 2 covers operating experience with the Dresden, Big Rock Point, and Humboldt Bay reactors. References 3 and 4 discuss preliminary operating experience with the Russian pressurized-water reactor, WWER. Operating experience with Shippingport, Yankee, the CETR, and Dresden has been reviewed in past issues of *Power Reactor Technology*, as indicated in Table II-2. Much of the information contained in Refs. 1 and 2 on Shippingport, Yankee, Dresden, and the CETR has been covered in these reviews, but Ref. 2 also contains significant information on the Big Rock Point and Humboldt Bay reactors. Both of these reactors went into commercial operation in late 1963 and have had

availability factors of about 95% through the end of the year, a period which did not involve refueling. Both reactors experienced leakage past the inner O-ring seal of the pressure-vessel head. In the case of the Big Rock Point reactor, this was attributed to a number of thermal cycles plus the use of three different operating pressures, and the leakage was stopped by replacement of the O-ring. The vessel-head flange-sealing surface of the Humboldt Bay reactor was modified by grinding to prevent further leakage.

An interesting repair technique was employed on the Big Rock Point reactor. Flow-induced vibration was found to be responsible for the loosening of a cap screw from one of the Zircaloy fuel channels; this vibration was caused by direct impingement of the recirculating flow on a portion of the core support, and the core-support plate had to be removed for installation of a new flow distributor. This work was performed inside the water-filled reactor vessel by four divers with a significantly low resultant radiation exposure.

References 3 and 4 discuss physics studies and preliminary experience with the Russian WWER. As of the date of the references (May 1964), the WWER had not produced nuclear electricity, although the turbines had been rolled on steam from auxiliary boilers. Critical experiments had been done, and the primary coolant system had been debugged.

Significant Developments

Fuel Elements, Fuel Materials, and Cladding

As indicated in Table II-1, the water-moderated reactors typically employ fuel in the oxide form (UO_2 , or a UO_2 - ThO_2 mixture as in the CETR and the ERR), usually as pressed and

Table II-1 BOILING- AND PRESSURIZED-WATER REACTOR-DESIGN DATA

Installation	Power, Mw(e)	Startup	Reactor pressure, psi	Steam conditions, psia/°F	Fuel/cladding	Reactor control	Remarks
Boiling-Water Reactors							
Dresden Nuclear Power Station, Morris, Ill.	200	1959	1015	Primary: 975/452 Secondary: 500/467	UO ₂ /S.S./Zircaloy-2	Control rods	
Rheinisch-Westfälisches Elektrizitätswerk (RWE), Kahl, West Germany	15.6	1960	1000	750/550	UO ₂ /Zr	Rods and solid shims	
Elk River Reactor (ERR), Elk River, Minn.	20.0	1962	937	865/900	UO ₂ ThO ₂ /S.S.	Rods and fixed (solid) shims	Indirect cycle
Big Rock Point Nuclear Plant, Charlevoix, Mich.	47.8	1962	1050	Direct cycle	UO ₂ /S.S.	Rods and fixed (solid) shims	Steam drum
Humboldt Bay Power Plant, Unit No. 3, Humboldt Bay, Calif.	50	1963	1035	Direct cycle	UO ₂ /S.S.	Rods and poison curtains	Pressure suppression
Japan Atomic Energy Research Institute Reactor (JPDR), Tokai- Mura, Japan	11.7	1963	905	495/530	UO ₂ /Zr	Rods and poison curtains	
Società Elettrolucare Nazionale Reactor (SENN), Punta Fiume, Italy	150	1963	1000	1000/(sat.)	UO ₂ /S.S. and Zr	Control rods and solid shims	Dual cycle, steam drum
La Crosse Boiling Water Reactor (LACBWR), La Crosse, Wis.	50	1965	1300	1300/577	UO ₂ /S.S.		
Kernkraftwerk-RWE-Bayernwerk Reactor (KRB), Gundremmingen, West Germany	237	1966	1015	500/467	UO ₂ /S.S.		
Indian Department of Atomic Energy BWR, Tarapur, India	380 (two reactors)	1968	1015	1015 primary/(sat.) 500 secondary/ (sat.)	UO ₂ /Zircaloy-2	Control rods	Dual cycle forced circulation, internal steam separation, pressure suppression
Oyster Creek Nuclear Electric Generating Station, Oyster Creek, N. J.	540	1967	1000	Direct cycle	UO ₂ /Zr		
Pacific Gas & Electric Bodega Bay Reactor, Bodega Bay, Calif.	313	1968	1075	Direct cycle	UO ₂ /S.S.		Pressure suppression
Nine Mile Point Nuclear Power Station, N. Y.	525	1968	1000	Direct cycle	UO ₂ /Zr	Rods and re- circulation pumps	Pressure suppression
N. V. Samenwerkende Elec- triciteits Productbedrijven (SEP), Netherlands	50	1968					

Pressurized-Water Reactors

Shippingport PWR, Shippingport, Pa.	60	1957	2000	600/486	U-Zr plates and UO ₂ /Zircaloy-2 rods	Rods and burnable poison
Centre d'Études de l'Énergie Nucléaire Reactor (BR-3), Mol, Belgium	10.7	1962	2000	520/471	UO ₂ /S.S.	Rods and chemical shim
Yankee Atomic Electric Co. Reactor, Rowe, Mass.	160	1960	2300 (max.)	465/460	UO ₂ /S.S.	Rods and solid shims
Consolidated Edison Thorium Reactor (CETR), Indian Point, N. Y.	255	1962	1500	370/1000	UO ₂ /ThO ₂ /S.S.	Control rods and solid shims
Novo-Voronezh Atomic Power Station (VVER), Russia	210	1964	1422	455/(sat.)	UO ₂ /Zr-Nb	Oil-fired superheater
Saxton Nuclear Experimental Corporation Reactor, Saxton, Pa.	5	1962	2000	475/462	UO ₂ /S.S.	Movable fuel and flux trap, safety rods
Società Elettronucleare Italiana (SELNI), Trino, Italy	240	1964	2015	500/467	UO ₂ /S.S.	Rods and chemical shim
Southern California Edison Co. San Onofre Reactor, Camp Pendleton, Calif.	375	1966	2065	650/495	UO ₂ /S.S.	
Connecticut Yankee Atomic Power Co. Reactor, Haddam Neck, Conn.	463	1967	2065	690/502	UO ₂ /S.S.	
Malibu Nuclear Power Station (Malibu-1), Corral Beach, Calif.	462	1968	2065	690/410	UO ₂ /S.S. or Zr	
Société d'Énergie Nucléaire Franco-Belge des Ardennes (SENA), Chooz, France	242	1966	2100	465/460	UO ₂ /S.S.	Control rods and chemical shim
Kernkraft Baden-Württemberg Planungsgesellschaft mbH, (KBWP), Obrigheim on the Neckar, West Germany	283	1968	2062	711/(sat.)	UO ₂ /Zr	Underground containment, in-core instrumentation
Union Electrica Madrileña Reactor, Zorita de los Canes, Spain	140	1968				Double containment

Table II-2 LOCATIONS OF PREVIOUS REVIEWS OF DESIGN AND OPERATING EXPERIENCE FOR WATER-MODERATED REACTORS IN POWER REACTOR TECHNOLOGY

Reactor	Item	Vol.	Issue	Pages
Shippingport	First refueling	4	2	52-58
	Power-generation history, fuel-element examination, experience with major equipment, radiation control, second core	5	4	61-72
	Instrumentation, second refueling	6	2	64-67
	FEDAL system, loss-of-flow experiments, radiation, hydraulics experiments, fuel failure	6	3	49-55
	Operating history, pressure-vessel irradiation, fuel examinations	7	2	193-196
	Design	4	3	47-55
Yankee	Head closure, containment experience, turbine vibration	4	4	69-70
	Control-rod examination	6	3	55-56
	In-core instrumentation, radioactivity problems, reactivity characteristics	6	4	82-91
	Design	4	4	56-68
Dresden	Control rods, grid-plate cracks, load following	5	2	56-59
	Water chemistry, load schedules, transient experiments	6	2	60-64
	Design	6	3	28-40
CETR	Fuel pool experience, control-system operation, performance	7	3	313-315
	Design	5	2	33-47
Elk River	Design	7	1	70-84
Humboldt Bay and Big Rock Point	Design	7	1	70-84

sintered cylindrical pellets in jackets of Zircaloy or stainless steel. An early exception was the Experimental Boiling Water Reactor, which used plates of uranium metal alloyed with small percentages of zirconium and niobium, clad with Zircaloy. The Shippingport reactor used Zircaloy-clad plates of uranium-zirconium alloy for the seed loadings installed in core 1; since the seed fuel is highly enriched, this alloy was rich in zirconium.

In Ref. 5 the results of the U. S. program on oxide fuels are summarized as follows:

During the past five years in the United States there has been a systematic approach to design formulation, fabrication development, and performance testing of uranium oxide type fuels. Sufficient data are available to allow a quantitative evaluation of oxide fuels for use in economic central-station power applications. The extensive scope of this program includes:

a. Over 130,000 Zircaloy and stainless steel clad UO_2 fuel rods have been irradiated in operating water and steam cooled (260°-316°C) power reactors. Maximum exposure attained is about 28,000 MWD/TU; maximum heat fluxes are about 350,000 Btu/hr-ft² for Zircaloy clad fuel and 450,000 for stainless clad fuel...

b. Over 4000 developmental fuel rods representing 37 different fabrication processes have been irradiated in water and steam cooled test reactors and loops. Maximum exposure attained was 16,000

MWD/TU and 1,435,000 Btu/hr-ft² peak heat flux. Plate UO_2 fuel has been irradiated in test loops to 127,000 MWD/TU exposure.

c. More recently, oxide fuel elements have been irradiated to 3600 MWD/TU maximum at 445,000 Btu/hr-ft² heat flux in superheat steam (510°C) test reactor loops.

The reference continues with a discussion of fuel-fabrication techniques and the behavior of fuel elements clad in zirconium alloys and stainless-steel alloys. Table II-3 gives additional details on oxide cores clad with Zircaloy-2 and stainless steel, which reflect U. S. design practice.

In general, the in-reactor behavior of the Zircaloy-clad elements has been "outstanding,"⁵ with but few failures. Purposely defected elements have operated relatively successfully, with little loss of UO_2 to the coolant and little reaction between the fuel and the available oxygen in the water; washout from powder-filled fuel elements has been slight. Hydrogen pickup by the zirconium-base alloys has generally proved acceptable, with in-pile pickup well correlated with autoclave corrosion data (Fig. II-1).

The outstanding example of in-pile experience with stainless-steel-clad fuel elements is

Table II-3 DETAILS OF POWER-REACTOR CORES FABRICATED WITH ZIRCALOY-2 AND STAINLESS STEEL⁵

	Power, Mw(e)	No. of rods	Rod outside diameter, in.	Cladding thickness, in.	Peak heat flux $\times 10^{-5}$, Btu/(hr)(sq ft)	Average exposure,* Mwd/ton of U
Zircaloy-Clad Fuel Elements						
Boiling-water reactors						
Dresden	220	15,500	0.570	0.030	3.46	9,000
Dresden†		192	0.470	0.025	3.46	7,000
Japan	12	3,460	0.564	0.033	2.96	1,000
Kahl	16	3,200	0.571	0.033	3.38	4,500
SENN	150	17,000				
Pressurized-water reactors						
Shippingport core‡	68	16,400	0.411	0.023	3.43	10,000
Shippingport core§	150	Flat plates	Flat plate, 0.100 in. thick	0.020	6.50	
Carolinas-Virginia¶	17	1,230	0.490	0.022	4.69	1,000
Stainless-Steel-Clad Fuel Elements						
Boiling-water reactors						
Dresden	220	4,900	0.440	0.019 (304 S.S.)	2.76	4,000
Elk River	20		0.452	0.020 (304 S.S.)	3.13	500
Humboldt Bay	65	8,428	0.463	0.019 (304 S.S.)	3.65	3,000
Big Rock Point	50	7,392	0.388	0.019 (304 S.S.)	3.50	3,000
Pressurized-water reactors						
Yankee	185	23,142	0.340	0.021 (348 S.S.)	4.46	8,000
Saxton	20 (thermal)	1,512	0.391	0.015 (304 S.S.)	4.44	3,000
Indian Point**	151	23,400	0.304	0.021 (304 S.S.)	5.40	3,000
N.S. Savannah	70 (thermal)	5,248	0.500	0.035 (304 S.S.)	2.77	1,500
SELNI	270	24,960	0.385	0.015 (304 S.S.)		Under construction

*Editor's Note: This is the exposure attained as of the writing of Ref. 5 and is not necessarily the maximum attainable.

†Straight-through 9-ft-long rods.

‡Rod fuel only in blanket region.

§Flat plates in entire core. Seed is 0.040-in.-thick plates with ZrO_2-VO_2 .

¶Zircaloy-4 cladding; this reactor is cooled and moderated by D_2O .

**Net power is 255 Mw(e), of which 104 Mw(e) is from fossil superheaters.

probably the Yankee reactor. The second core of this power reactor contained two assemblies from the first core, and these two assemblies operated satisfactorily to a maximum burnup of 27,000 Mwd per ton of uranium. Detailed examination of the first core showed that the cladding exhibited no microstructural changes, an increase in burst strength, "limited" amount of crud, and no significant structural changes.⁵ Thin-walled cladding, relying on the fuel for support, has been tested in Dresden and in the Vallecitos Boiling Water Reactor (VBWR) with less success; the VBWR results were reviewed in detail in the Winter 1963-1964 issue of *Power Reactor Technology*, 7(1): 27-50, and are summarized in Fig. II-2.

Pertinent conclusions from Ref. 5 are as follows:

Zircaloy-2 clad rod fuel has been operated satisfactorily to maximum burnup of 16,000 MWD/TU at peak heat fluxes of 510,000 Btu/hr-ft², and tests are continuing.

Heavy walled "free standing" fully annealed stainless steel rods have operated satisfactorily to 27,000 MWD/TU maximum and peak heat fluxes of 446,000 Btu/hr-ft².

Thin clad "nonfree standing" stainless has shown inter-granular stress induced corrosion at average burnups less than 10,000 MWD/TU and high heat fluxes. Exact cause of the failure mechanism is still under investigation.

It should be noted, however, that more than one variable was changed between the freestanding- and nonfreestanding-cladding experiments. For example, the heavy-walled freestanding fuel elements were irradiated in a pressurized-water-reactor (PWR) environment and were fabricated of type 348 stainless steel. The thin nonfreestanding claddings were irradiated in Dresden and VBWR in a boiling-water-reactor (BWR) environment and were fabricated of type 304 stainless steel.

The irradiation behavior of ceramic fuels is discussed in Ref. 6. The paper presents engineering type information on a number of fuel

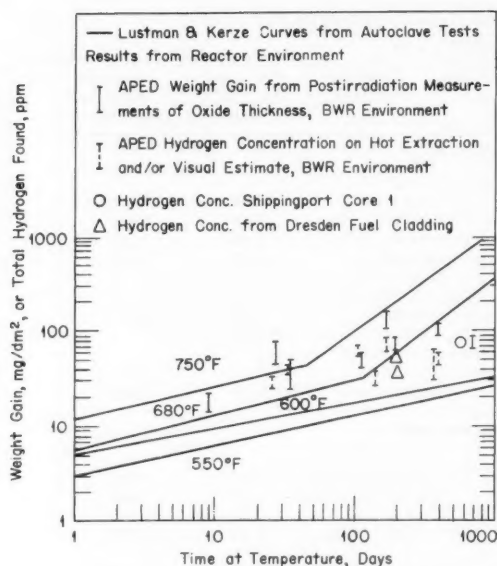


Fig. II-1 Corrosion behavior and hydrogen pickup of Zircaloy-2. Comparison of autoclave and reactor exposures.⁵

materials, including UO_2 , $\text{ThO}_2\text{-UO}_2$, $\text{PuO}_2\text{-UO}_2$, ZrO_2 base- UO_2 , $\text{UO}_2\text{-BeO}$, $\text{UO}_2\text{-Al}_2\text{O}_3$, MgO-PuO_2 , $\text{ZrO}_2\text{-PuO}_2$, UC, PuC, UC-PuC, UN, US, and U_3Si . Of these materials the oxides are of primary importance to water reactors and, naturally, have been investigated most extensively. For example, the thermal conductivity of UO_2 , illustrated in Fig. II-3, appears to be reasonably well determined. Increases in k_{UO_2} at high temperatures, where it has been postulated that radiative-energy-transport mechanisms become important, have not been observed in polycrystalline UO_2 even after columnar grain growth. Fission-gas release, although not well understood, is amenable to engineering calculations, at least to the accuracy required for the practical design of fuel elements for the H_2O reactors.* Structural changes in UO_2 have been quantitatively related to fuel operating temperature, temperature

*An extensive discussion of the engineering and performance of UO_2 fuel assemblies was presented by the Canadians in Refs. 30 to 32. Their special interest in fission-gas release presumably comes from the fact that they cannot design NPD type fuel rods with large fission-gas spaces to allow for imperfections in gas-release calculations without introducing additional difficulties related to end-cap peaking, fuel shuffling, etc.

gradient, exposure, density, and irradiation history. Reference 6 suggests that future reactor designs may employ lower density UO_2 to encourage fission-gas release and reduce swelling during irradiation. This may be of importance in the design of oxide fuels for long-term irradiation in fast reactors or for those reactors where the provision of a fission-gas plenum results in an unacceptable performance penalty. Reference 7 reports preliminary results for fabrication and irradiation of $\text{UO}_2\text{-PuO}_2$ fuel elements in the BR-3 reactor.

References 8 and 9 report on various experiments designed to study the corrosion resistance of zirconium alloys. Reference 8, which is primarily of interest to the specialist, is on metallurgical techniques in which radioactive tracers are used for the study of zirconium alloy corrosion; it also presents some results. Reference 9 is devoted to the zirconium

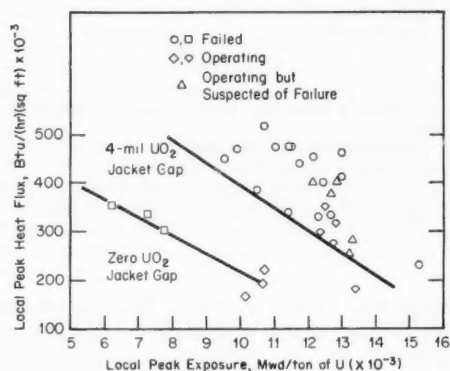


Fig. II-2 Local peak heat flux (within 500 to 1000 hr of failure) vs. local peak exposure for 304 stainless-steel-clad UO_2 fuel.⁵

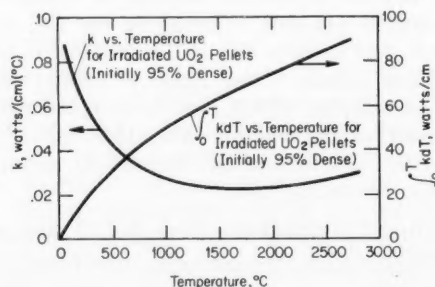


Fig. II-3 UO_2 thermal conductivity and $\int_0^T kT$ vs. temperature.⁶

nium-niobium alloy hydriding problem in water-cooled reactors. This class of alloys is an interesting one since the Russians have traditionally used the zirconium-niobium alloys in their water-cooled power reactors, whereas the Zircalloys have been used almost exclusively in the United States. The reference discusses the effect of stress upon the distribution, or arrangement, of the zirconium hydride corrosion product. The hydride is in the form of platelets, and, if these are oriented parallel to the axis of the tube, their effect on tensile properties is minimized. On the other hand, if the platelets are radially oriented, serious reductions in tensile strength occur. According to Ref. 9:

...formation of radially oriented hydrides is a function of values of the tensile stress and does not depend upon the methods of hydriding. Even at the hydrogen content of 0.01% with radially oriented hydrides the pipe properties are sharply impaired; that proves the paramount importance of the values of tensile stress when using zirconium alloys in water-moderated water-cooled power reactors. On all other conditions being equal, preference should be given to the zirconium alloys possessing high yield point as in this case the value of the allowable hydriding stress will be higher and reliability of operation will correspondingly increase.

The allowable hydriding stress of the alloy is the maximum tensile stress at which radial orientation of the hydrides under the specified service conditions does not yet occur. Evidently, it is to be taken into account along with other parameters in designing zirconium alloy claddings for fuel elements or zirconium alloy pipes for service under pressure.

Values of the allowable hydriding stress for two of the zirconium-niobium alloys are given in the reference. For zirconium-1 wt.% niobium, radial orientation of the hydrides occurred at 4.4 kg/mm²; and, for zirconium-2 wt.% niobium, radially oriented hydrides were formed at a stress of 8.8 kg/mm² and over. These values are about half of the yield strength of each alloy at the test temperature of 400°C. Although the stronger zirconium-niobium alloys have not excited much interest in the United States as cladding materials, their use in pressure-tube type reactors and for in-core structural components of pressure-vessel type reactors may increase.

Reference 10 summarizes a considerable amount of U. S. experience on corrosion in aqueous systems. The generally successful use of Zircaloy and stainless steel in power reactors is confirmed, and the superiority of beta-

quenched Zircaloy-4 over Zircaloy-2 in high-temperature water is discussed. With regard to stainless steel, the authors note that "...highly stressed thin wall cladding, representing advanced fuel concepts, has failed in VBWR after exposures considerably shorter than required for economic performance..."¹⁰ Although this is true, failures have also been noted in VBWR irradiation of types H and I elements in the Fuel-Cycle Program; these elements were designed to be freestanding [see reviews in *Power Reactor Technology*, 7(1): 27-50 and 7(4): 354-355].

In summary, there is little doubt that metal-jacketed oxide fuel elements can be made to have attractively long service lifetimes in water-cooled reactors. The quantitative principles of optimum design are not completely understood, however, and caution is in order in making substantial departures from the designs which have been proved out by reactor experience. It follows also that, in quoting and using the experience obtained to date, one must be particular and comprehensive in considering the conditions under which the experience was obtained. Some departure from this principle was evident in the informal discussion in the Geneva sessions, where the Shippingport experience with exposure lifetime of the blanket elements appeared to be regarded as typical for oxide fuel elements, even though the thermal demands on the oxide in these elements are relatively quite mild [see the review of Shippingport Operating Experience in *Power Reactor Technology*, 7(2): 193].

Water Chemistry: Use of Mild Steels in Aqueous Systems

The use of mild steels in portions of power-reactor primary circuit piping has been common practice for some time; Dresden, for example, uses low-alloy and carbon steels for the primary and secondary steam piping and the feedwater system. A full-flow condensate demineralizer is provided, however. In Ref. 10 the use of boiler steel is considered for unclad pressure vessels, and the conclusion reached is that the question is "unresolved." The problem is of more than academic interest, however, since at least one power reactor under construction (the second unit of the WWER) is said²⁹ to utilize an unclad pressure vessel. Accordingly, the discussion presented in Ref.

11 serves to illuminate some of the problems associated with the use of mild steel in power-reactor primary circuits.

The water and steam of a typical boiling-water reactor will contain about 0.1 and 20 ppm oxygen, respectively.¹¹ Experiments done at the Kjeller Research Establishment in Norway, and summarized in Ref. 11, indicate that the initial corrosion-product release rate under these conditions is "high" but, under continued high-temperature operation, decreases to a release rate as low as 2 mg/(dm²)(month). This is comparable to values for stainless steel. The situation has additional complications in the case of reactor operation since the system may from time to time be kept at lower temperatures for relatively long periods (as during refueling); rather high corrosion rates have been reported in neutral water at low temperatures. The paper¹¹ suggests that during shutdown the oxygen content of the water should be monitored and reduced to the point that harmful corrosion will be prevented. Under low-oxygen conditions the corrosion film formed at high temperatures is apparently stable, although for long periods of downtime it may be necessary to scavenge the oxygen.

As a further consequence of the corrosion in neutral water at low temperature, low-temperature portions of the primary system may present difficulties in the absence of condensate cleanup systems. The corrosion products introduced into the reactor may precipitate on fuel elements, and Ref. 11 mentions that "rust generators" are to be installed in the Halden Boiling Water Reactor (HBWR) to study the problem.

Control: Soluble and Burnable Poisons

The use of soluble poisons to control reactivity in pressurized-water reactors was reviewed in detail in Sec. IX of *Power Reactor Technology*, 7(4). References 12 to 14 are on aspects of soluble-poison control, and the information reviewed here will be supplementary to the material presented in the 7(4) issue.

The dynamic performance of a chemical-shim system is the subject in Ref. 12. The reactor that was used as a neutron source was a swimming-pool type, and the boric acid was contained in a pipe introduced within the core. The control system (feed and bleed) and associated piping and electronics probably are

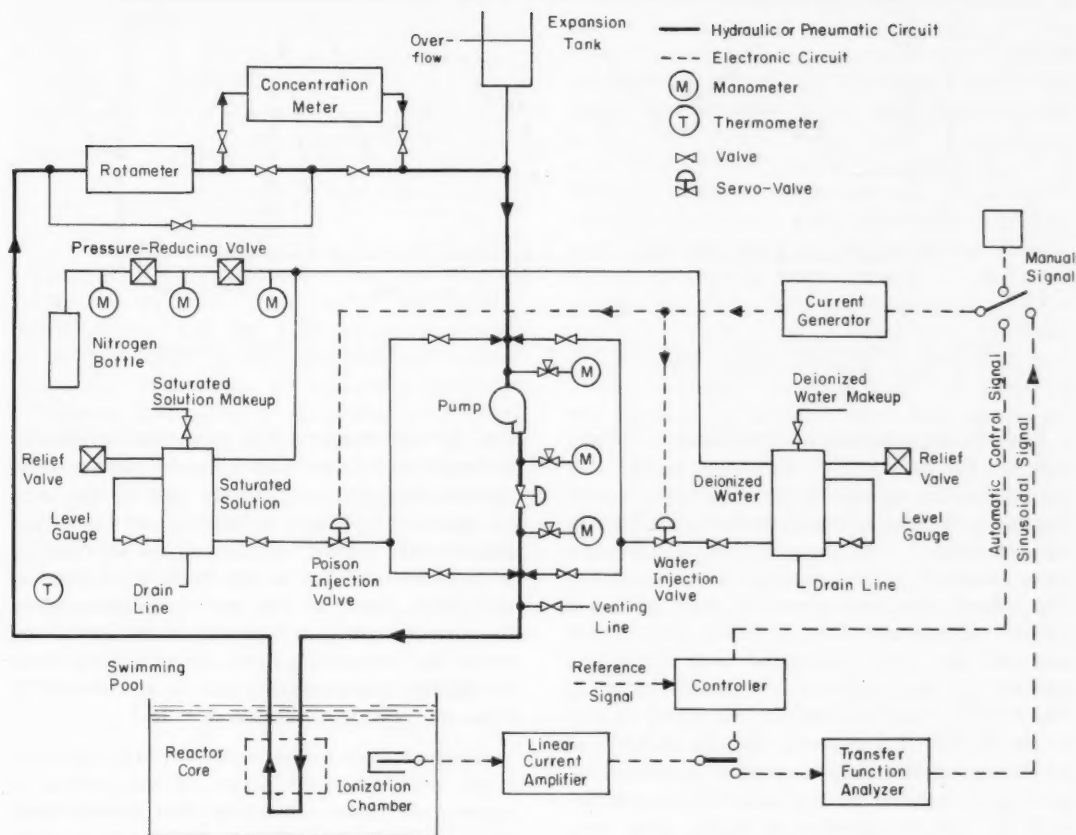
representative of what could be used for power-reactor control, but the range of dynamic variables was much greater than that encompassed by the chemical-shim application currently considered in power reactors. The response of the control system to various reactivity steps and ramps was investigated with the following results:¹²

...the control system examined here seems to have satisfactory dynamic characteristics as regards the fine control of the neutron flux level. There are no particular operation and maintenance problems. The whole system appears to be quite reliable, rugged and easy to handle.

The flow diagram of the poison loop is shown in Fig. II-4.

Operation of the Belgian BR-3 plant with boron in the coolant is described in Ref. 13. The steps performed to prepare the system for operation with the chemical shim, and subsequent transient tests, are detailed in Table II-4. The physical, chemical, and radiochemical characteristics of the reactor coolant were monitored during the course of operations in the chemical-shim experiment, which lasted about a month. In general, the results of the experiment were good. Small, unexplained, reactivity variations occurred (Fig. II-5), but they happened over relatively long times and, in fact, were noticed in normal operations without boron. Water chemistry showed no unusual or unexpected trends, and the reactor coolant activity remained below design limits, although the demineralizer has been bypassed for more than six weeks.

Reference 13 concludes with an interesting discussion of the application of chemical shim to a large [250 Mw(e)] reactor. Depending upon the operational program, from 2 to 10 tons of boric acid per year will have to be disposed of. If the core has contained failed fuel elements, this material will be contaminated by fission products to a varying degree, depending on the number of failed fuel rods. Distillation, electrodialysis, and ion exchange are considered as concentration processes, but no specific recommendations are given. It may be that the success of the soluble-poison concept relies on keeping the number of routinely failed fuel elements within some limit, and the value 1% is mentioned. Fortunately, the experience with the reactor type has illustrated that this limit is probably not an unrealistically difficult one.

Fig. II-4 Simplified flow diagram of soluble-poison control system.¹²Table II-4 OPERATIONS WITH THE BR-3 REACTOR: CHEMICAL-SHIM EXPERIMENT¹³

1. Preliminary operation at normal power level, without the mixed-bed demineralizer; the ion beds are not saturated in BO_3^- ions and therefore cannot be used during the boron test.
 2. Preliminary operation at normal power level, without demineralizer, the pH being reduced from 9.5 to approximately 7.
- During these two preliminary tests, the increase of the reactor coolant activity (very predictable) was, however, far from the permissible limits.* Also the air contamination remained, everywhere, far under the maximum permissible concentrations.
3. Power operation with 125 ppm boron present in the primary coolant. This first step in the study of chemical control made it possible to determine the overall effect of bypassing the demineralizer and of the presence of boron in the reactor coolant. It gave information (a) on the evolution of chemical characteristics of the water, the boron concentration, and the core reactivity and (b) on the possibility of compensating the xenon poisoning.
 4. Power operation with a higher boron concentration in the reactor coolant water (350 ppm). This last step, performed after a core shuffling, made possible a more extensive study:
 - Startup of the plant, and load pickup with boron present
 - Overall compensation of the xenon effect by reduction of the boron concentration
 - Power operation with all rods extracted from the core, except two rods for fine control
 - Compensation of samarium poisoning and fuel burnup by boron removal. Furthermore, various load rejections and fast load pickups made it possible to test the dynamic behavior of the reactor.

*The activity of the primary coolant increased about a factor of 3 when the resin-bed purification system was not operational. The new value¹³ was about 0.24 $\mu\text{C}/\text{ml}$.

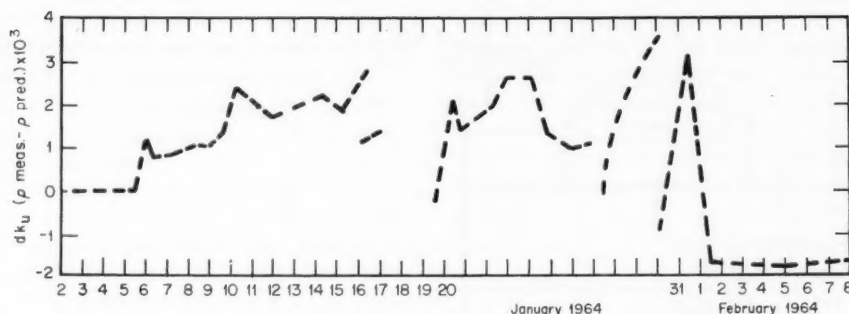


Fig. II-5 Unaccounted reactivity disappearance, dk_u , vs. time. Results of power operation of the BR-3 reactor in the presence of 400 ppm boron.¹³

Operational experience with Yankee, Saxton, and the BR-3 is briefly discussed in Ref. 14. An interesting result of the experiments with Yankee and Saxton has been the discovery of the "pH effect"—an increase in reactivity has been observed with increase of the coolant pH. The effect has been observed both with and without significant boron levels in the coolant and has not been correlated with the poison content or corrosion-product concentration in the coolant. The increase in reactivity is only a few tenths of a percent, but, as of the time of writing of Ref. 14, no satisfactory explanation had been offered. Hot-laboratory examination of core components of Saxton after nine months of operation with dissolved poison showed "trivial quantities of residual poison on the core."¹⁴

Reference 14 also discusses recent advances in burnable-poison control. Some of the discussion centers around core 2 of the Shippingport PWR, which utilizes boron-stainless-steel strips as a burnable poison (Fig. II-6). This

type of reactor appears to be particularly suited to burnable-poison control since (1) the burnable poison need be incorporated only in the seed elements, (2) only a relatively few burnable-poison sites need be employed, and (3) there is no complication due to the shuffling of fuel assemblies. Also, in the seed-blanket reactor, the control rods are placed in the seed, and hence the burnable poison can be located so as to reduce local peaking due to rod channels. This approach is evident in Fig. II-6.

Although most of the burnable-poison applications have specified boron as the poison, a number of other materials offer possibilities for use. These are members of the rare-earth family (samarium, dysprosium, erbium, and gadolinium), and each may offer an advantage relative to boron in a particular situation.

Pressure Vessels

Three papers on reactor pressure-vessel design and fabrication¹⁵⁻¹⁷ are from three different countries and present dissimilar views on many aspects of pressure-vessel technology. For example, the "standard" material used for large pressure vessels in the United States is SA-302 grade B steel, the material used for the SEP and SENA pressure vessels is a steel designated "1,2 MD07," and the material of the WWER vessel is a high-strength Cr-Mo-V steel. The exact composition of the latter was not specified, but the compositions of the other two are given in Table II-5. The selection of a pressure-vessel material usually involves a pressure-vessel code, and Ref. 15 contains a discussion of the effects of pressure-vessel codes of various countries.

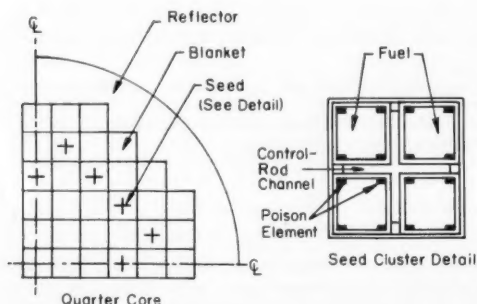


Fig. II-6 Schematic arrangement of burnable poison in the Shippingport core geometry.¹⁴

Table II-5 COMPOSITIONS OF VARIOUS PRESSURE-VESSEL STEELS

Country	Designation	Nominal composition, wt.%
United States	SA-302 grade B	<0.25 C, 1.15-1.50 Mn, 0.45-0.60 Mo
Netherlands	1,2 MD07	<0.18 C, 1.0-1.5 Mn, <0.2 Cr, 0.3-0.7 Ni, 0.35-0.55 Mo

In the fabrication of the Russian- and Netherlands-built pressure vessels for the WWER, SENN, SENA, and the KRB reactors, forged rings were used to form the pressure-vessel bodies.^{15,17} In contrast, the cylindrical portions of vessels fabricated in the United States are usually rolled, and, if the vessel wall is thin enough, the cladding is roll bonded. The advantages of the forged-ring fabrication technique are discussed in Ref. 15, although, as pointed out in that paper, the resulting vessel is higher in price than a similar one built of sheet material.

Pressure-vessel design and fabrication in the United States are discussed in Ref. 16. The reference opens with a discussion of the recent amendments to the ASME *Pressure Vessel Code* which have resulted in the approval of Sec. III of the ASME *Power Boiler and Unfired Pressure Vessel Codes*, "Rule for Construction of Nuclear Vessels." Older vessels were fabricated in conformity with either Sec. I or Sec. VIII. Although much of the discussion is primarily of interest to the pressure-vessel specialist, Fig. II-7, which illustrates typical BWR and PWR pressure vessels fabricated by either of the ASME codes, Sec. III or Sec. VIII, is a graphic summary of some important effects of the changes. In the particular cases shown, the design by Sec. III results in significant changes in wall thicknesses for the cylindrical sections of the vessel; Table II-6 gives an estimate of the effect on fabrication time for BWR and PWR vessels of various sizes. At first consideration it is surprising that the fabrication times for vessels for both PWR and BWR ap-

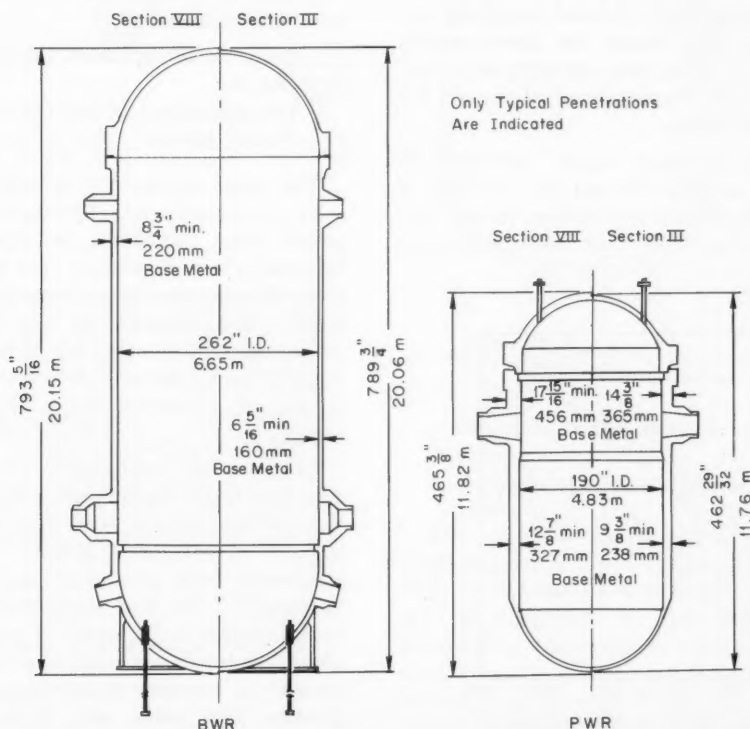


Fig. II-7 Reactor vessel designs for 1000-Mw(e)-capacity plants comparing ASME Code Secs. III and VIII.¹⁶

Table II-6 FABRICATION TIME FOR REACTOR VESSELS OF VARIOUS SIZES¹⁶

Plant capacity, Mw(e) net	Fabrication time, weeks	
	Sec. III design	Sec. VIII design
250	95	100
500	120	125
750	145	155
1000	165	175
1500	205	220

plication are approximately equal. The reference explains that this is true because the combination of vessel-diameter and vessel-design-pressure requirements offset each other. This, of course, may not always be the case. The relative size difference between BWR and PWR vessels, as indicated in Fig. II-7, is striking. Figure II-8, from the same paper, presents the data in slightly different form. The vessel-size requirements are, of course, determined by reactor design, rather than by vessel-design considerations. Yet the plotted points, although they are not identified in the paper, presumably represent vessels specified for actual plant designs. Hence the lower section of the curve, at least, has authority as a representation of the current state of the art for BWR and PWR design.

The WWER pressure vessel described in Ref. 17 is an unusual one and, for purposes of description, is divided into lower, middle, and upper zones (Fig. II-9). The lower zone con-

sists of three ring portions, each forged as a single piece. The middle zone contains 24 nozzles, which are welded in place as shown in Fig. II-9. The upper zone consists of a one-piece forged flange and a strengthening ring, fitted by means of a shrink-fit technique. The vessel cover is a forged flat plate with 55 penetrations. The internal surface of the vessel is covered with weld-deposited austenitic stainless steel having a thickness of about 20 mm—quite thick in terms of practice in the United States. The vessel-to-cover seal is accomplished by a wedge-shaped nickel gasket compressed between the cover and vessel by means of a thrust ring held in place by studs. The reference describes an alternative seal involving the welding of a flexible collar to both the vessel and the head.

Thermal and Hydraulic Design

The relatively large number of papers devoted to thermal and hydraulic considerations of water-cooled reactors were all concerned with boiling phenomena and could be grouped into three categories:

- Basic studies of bubble dynamics, velocity profiles, etc.
- Interpretations of experimental data
- Survey papers

The basic studies¹⁸⁻²⁰ of boiling phenomena were concerned primarily with bubble dynamics, active sites, nucleation, and other variables of interest in two-phase flow. The authors initiated their discussion with a consideration of pool boiling, and, although the detailed results are of primary interest to the specialist, they do indicate that continued progress is being made in gaining an understanding of two-phase behavior.

Of the papers concerned with experimental and analytical results, two presented information on the critical heat flux during unstable burnout in two-phase flow,^{21,22} and two were concerned with predicting ϕ_{BO} during stable burnout.^{23,24} The test loops used in the experiments reported in Refs. 21 and 22 both employed natural circulation and were instrumented to measure appropriate pressures, powers, flow rates, etc. Typical flow-power curves are illustrated in Fig. II-10. As shown, the flow initially increased, reached a maximum, and then decreased. As power was increased

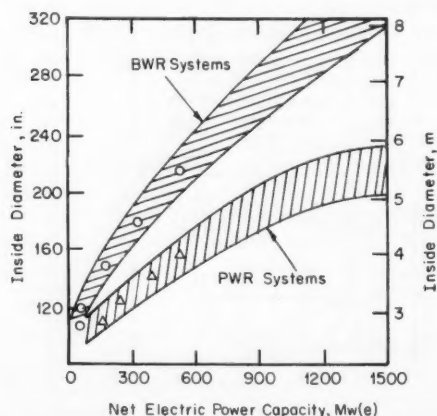


Fig. II-8 Reactor vessel diameter vs. plant power rating.¹⁶

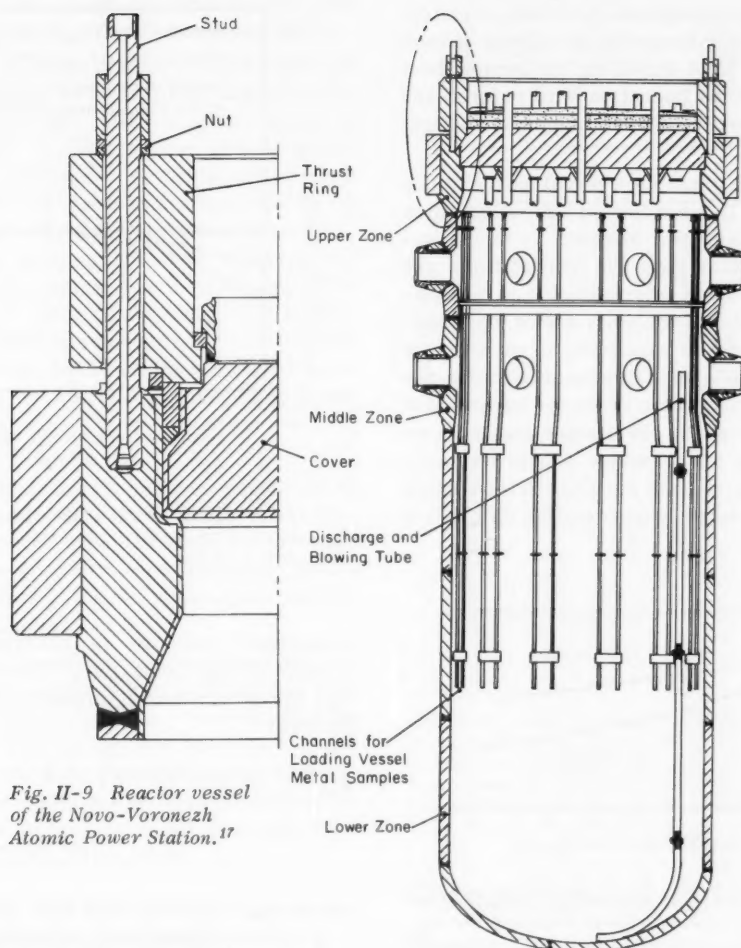


Fig. II-9 Reactor vessel of the Novo-Voronezh Atomic Power Station.¹⁷

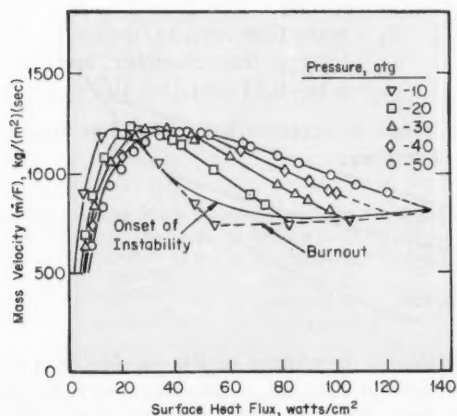


Fig. II-10 Effect of pressure on mass velocity for varying heat flux. Data taken in a natural-circulation loop with test-section diameter of 36 mm and length of 4890 mm. Inlet subcooling, $\Delta T_{\text{sub}} = 8 \pm 0.3^\circ\text{C}$.

in steps, oscillation usually commenced; three cases are reported:

- Diverging oscillations causing burnout
- Stable oscillations
- Burnout without oscillations

The experimental results were compared against predictions of a theoretical flow model. The predictions were obtained by numerically solving equations of continuity, energy, and momentum; the stability limit was analytically determined as the power input where the oscillations in inlet velocity became divergent upon a step increase of 1% in channel power. Although no general stability criteria are given in the reference, it would appear that the method is generally applicable for specific cases of uni-

formly heated test sections of simple geometry. Reference 21 also discusses the effects of inlet subcooling and inlet throttling on burnout heat flux and compares operation with natural and forced convection; these data are shown in Figs. II-11 to II-13.

Reference 23 presents several correlations of burnout heat flux for flow of subcooled water in tubes, steam-water mixture in tubes, and flow in an annular gap. The correlations are empirical, and their interpretation is complicated by the fact that the units are of the "mks" system. Pressure is measured in newtons per square centimeter and latent heat in joules per kilogram; the correlation gives the burnout heat flux in units of watts per square meter, however. The units for pressure should be carefully noted since at least one English translation of a recent Russian heat-transfer article has

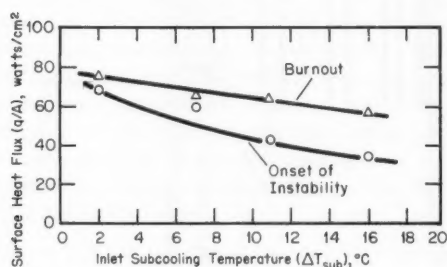


Fig. II-11 Effect of inlet subcooling on critical and burnout heat fluxes.²¹ Data taken in a natural-circulation loop with test-section diameter of 20 mm and length of 4890 mm. Pressure = 50 atg.

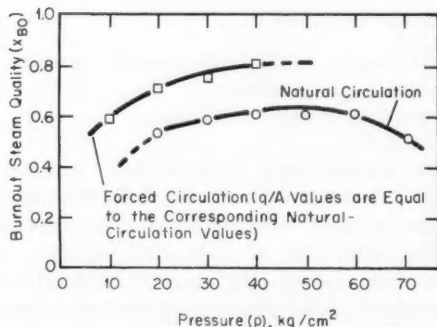


Fig. II-12 Comparison between natural- and forced-circulation burnout data.²¹ Test-section diameter, 20 mm; length, 4890 mm. Forced-circulation data were taken with "heavy" inlet throttling.

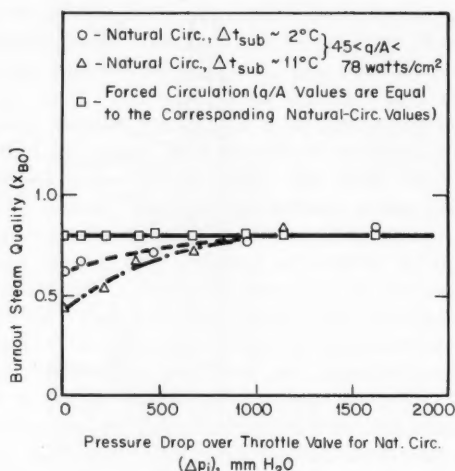


Fig. II-13 Effect of inlet throttling on burnout.²¹ Test-section diameter, 20 mm; length, 4890 mm. Pressure = 50 atg.

translated pressure in newtons per square centimeter as "neuts/cm²." The correlation for ϕ_{BO} for steam-water mixture in tubes is as follows:

$$\phi_{BO} = [1.46 \times 10^{-4} r^{1.72} (1-x)^m - 0.116 W_g] \frac{2.71}{d_{in} 0.48} \quad (1)$$

where ϕ_{BO} = burnout heat flux, watts/m²
 r = latent heat, joules/kg
 x = quality
 W_g = mass flow rate, kg/(hr)(m²)
 d_{in} = internal tube diameter, mm
 $m = 3.48 - 0.54 (r/4.18 \times 10^6)$

The range of applicability of Eq. 1 is stated to be as follows:

Pressure: 392–981 newtons/cm² (568 to 1422 psi)
 Mass flow: $2-18 \times 10^6$ kg/(hr)(m²)
 Quality: 0–0.40
 d_{in} : 4–12 mm
 Length: ≥ 200 mm

Equation 1 is plotted in Fig. II-14 for a pipe diameter of 0.18 in., a pressure of 1000 psia, and two different values of W_g . This figure compares the correlation with data appearing in Fig. III-12 of *Power Reactor Technology*, 7(1). The correlation does not predict the experimentally determined "best-fit" curve very

well, although the range of experimental data falls within the range of applicability of Eq. 1.

Reference 23 also presents the results of burnout experiments utilizing nonuniformly heated test sections, and these are given in Fig. II-15. The ordinate on the curves is believed by the reviewer to be the weighted mean heat flux, and the heat-flux nonuniformity is believed to be a radial one, although the reference is not particularly clear in these two quite important aspects. The experiments were carried out "... with round tubes of $d_{in} = 10$ mm, tube length is 400 mm and with the eccentricity between external surfaces and internal ones (three values)..."²³ These three values are probably the ones corresponding to curves 2, 3, and 4 in Fig. II-15.

Reference 24 presents burnout data taken at General Electric—San Jose, Hanford, and Columbia. Since the results of many of these programs have been reviewed in previous issues of *Power Reactor Technology*, additional details will not be discussed here.

References 25 and 26 are in themselves reviews. Reference 25 reviews highlights of two-phase-flow research relative to boiling reactors in the United States, and Ref. 26 describes the experimental work on heat transfer and hydrodynamics with steam-water and two-component mixtures done at the Centro Informazioni Studi Esperienze (CISE), Milan, Italy.

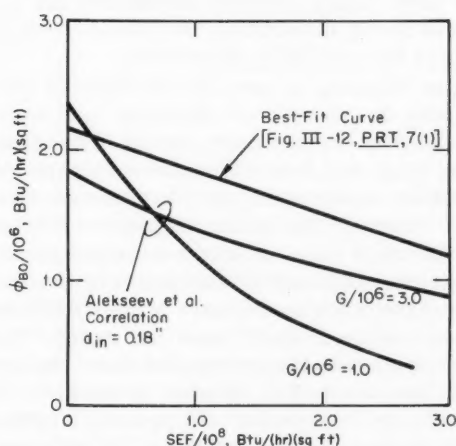


Fig. II-14 Comparison of burnout correlation of Alekseev et al.²³ with burnout data previously published in *Power Reactor Technology*. Pressure = 1000 psia.

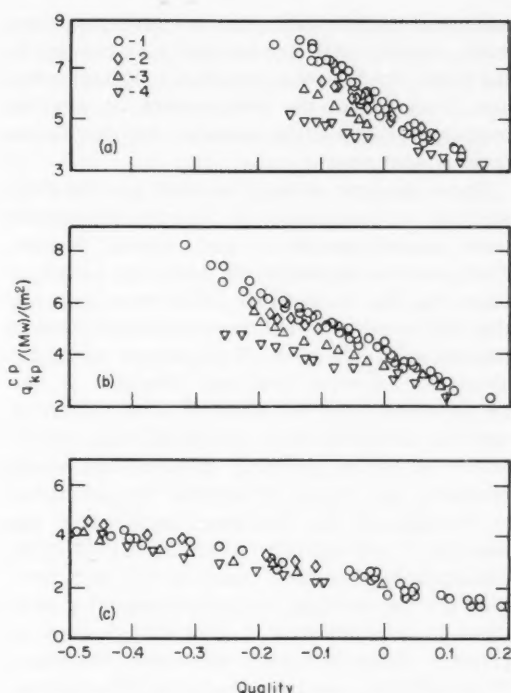


Fig. II-15 Heat-flux concentration nonuniformity influence on q_{BO} in round tubes. Curves 1, 2, 3, and 4 correspond to values q_{BO}^{max}/q_{BO}^{av} of 1, 1.12, 1.28, and 1.50. Parts a, b, and c correspond to pressures²³ of 588, 981, and 1765.8 neutrons/cm².

Directions of Future Developments

To some extent the future developments in water-reactor technology are tied into the design features of the advanced plants of each type; these are listed last under the appropriate headings in Table II-1. References 27 and 28 present views of two water-reactor manufacturers on developments of their particular type, and Ref. 29 discusses advanced Russian water reactors.

The boiling-water reactors designed by the General Electric Company are tied quite closely to the concept of pressure suppression. On the other hand, the recent Westinghouse-designed reactors utilize a variety of containments from an unshielded steel containment vessel to the double-barrier concept, employing a sandwich of porous concrete between two layers of steel and reinforced concrete. Both the pressure-

suppression and double-barrier concepts, however, require that the capability be present in the plant to process potentially contaminated air from within the containment or reactor building during routine operation before release from a plant stack.

Early designs of both the BWR and the PWR stations, as exemplified by Dresden and Yankee, used canned pumps in the primary circuit. This practice seems to be declining, however, since the Big Rock Point BWR uses pumps of the shaft-seal type and the San Onofre PWR is scheduled to use them. This change has come about for several reasons. Possibly in the pressurized-water reactors the most important was the desire to utilize pumps of capacities in excess of 50,000 gal/min. In the boiling-water reactors, the change is probably due primarily to confidence that has been built up in the handling of radioactivity: since the direct-cycle boiling-water reactors must, in any case, provide for the handling of radioactive off-gases, there is less incentive to reduce the leakage of primary water to a bare minimum. Reference 27, in addition, mentions that work is being done to develop a jet pump for use inside the reactor vessel of the boiling-water reactor to "substantially" reduce the flow external to the vessel. For other primary system components, Ref. 28 states that PWR plants currently under design and construction do not provide for primary-coolant-loop stop valves and check valves, since safety and operational studies indicated these valves were not needed.

Fuel-element experience has been relatively good for both the pressurized-water and boiling-water reactors, with the exception of the failures of the stainless-steel-clad elements in the VBWR mentioned previously. As a result, Ref. 27 states that Zircaloy is the "favored" cladding material for current boiling-water reactors, with Incoloy and Inconel slated for testing. On the other hand, the pressurized-water reactors have shown no difficulties when utilizing either stainless-steel cladding or Zircaloy, although large-scale experience with rod elements of the UO_2 -Zircaloy type has been limited to the rather mild conditions in the Shippingport blanket. Until the VBWR failures can be tied to a cause not connected with the reactor type, the PWR designer may have additional freedom in the choice of cladding materials because of the ability of the pressurized-water reactor to operate with close control of coolant

chemistry. Both references mention that spring type spacers will be considered for present and future fuel-assembly fabrication techniques. This eliminates welding and brazing operations on fuel-cladding surfaces and allows cold-worked cladding to be utilized, if desired.

Control and refueling are two areas of interest for future developments. The chemical-shim control of the pressurized-water reactor has been mentioned previously. In addition, Ref. 28 mentions a so-called "rod-cluster-control" scheme for advanced pressurized-water reactors, to be used in conjunction with chemical shim. This utilizes hollow tubes that replace some of the fuel rods within a fuel assembly; inside each hollow tube is a cylindrical, unfollowed, absorber rod. The several rods within the confines of a single fuel assembly are mechanically connected and are operated, as a cluster, by a single drive unit. No cruciform type control rods are employed. The application of soluble-poison control to the boiling-water reactor is less straightforward and has not been proposed in any commercial design. The use of bottom-entering control rods to assist in axial power flattening will no doubt be continued, and further flattening and reactivity compensation may be obtained with fixed shims of boron stainless steel. Both Refs. 27 and 28 mention "scatter loading," in which exposed fuel assemblies and new assemblies are distributed more or less uniformly over the core volume rather than in zones. Upon refueling, the most highly exposed assemblies are replaced by new assemblies, but there is no shuffling of the remaining assemblies.

The refueling of both reactor types is complicated by above-core structure and mechanism. The boiling-water reactor of the future most likely will have internal steam separators, and these will probably have to be removed before refueling. The pressurized-water reactor, on the other hand, while not requiring steam-separation equipment within the pressure vessel, may have control-rod-drive guide structures above the core which must be moved. This structure becomes more complex when "cluster" rods are used. The efficient removal of the above-core components will probably continue to be an important consideration for both reactor types in order to minimize downtime for refueling purposes. In-core instrumentation, mentioned in both references, will probably continue to follow the control-rod-drive posi-

tion, being inserted into the BWR core from the bottom and into the PWR core from the top. Thus the BWR in-core instrumentation need not be removed during fuel handling, whereas Ref. 28 mentions that the PWR in-core instrumentation probably will be removed as a unit, being attached to a common support plate presumably located above the core. One of the criticisms of the water reactors, heard rather frequently in the informal discussions at Geneva, was the long shutdown time required for refueling. No doubt this problem will receive increasing emphasis in coming years.

assembly consists of a guide tube mounted permanently in the core; its shape is hexahedral outside and cylindrical inside. The safety rod is fabricated of a cylindrical absorber 125 mm in diameter, which is followed by a zirconium alloy scatterer without fuel in the WWER-1 and by a fueled follower in the WWER-2.

Some of the differences between the two reactors—for example, the change to electric pressurization and the adoption of ion-exchange water treatment—involve changing to design techniques that have been used in the United

Table II-7 COMPARISON OF WWER-1 AND WWER-2

Parameter	WWER-1	WWER-2
Power, Mw(e)	210	365
Status	Operating	Under construction
No. of stationary fuel assemblies	312	276
Length, m	2.5	2.5
Shape, cross sectional	Hexagonal	Hexagonal
Distance across flats, mm	144	144
Pitch	147	
Fuel-pin geometry	Rods	Rods
Cladding material	Zr-Nb	Zr-Nb or S.S.
Cladding outside diameter, mm	10.2	8.8
No. of pins/assembly	90	127
Fuel material	UO ₂ pellets	UO ₂ pellets
Fuel-pellet outside diameter, mm	8.7	7.7
No. of movable fuel assemblies		
Compensating	31	60
Safety	6	13
Coolant purification	Evaporation	Ion exchange
No. of primary loops	6	3
Pressurizer type	Gas	Electric
Coolant pressure, atm abs.	100	120
No. of turbines	3	5

Reference 29 is interesting partly because the Voronezh reactors (WWER) are quite different in design from the pressurized-water reactors constructed in the United States. The reference discusses the changes between the first unit of the plant (WWER-1) and the second (WWER-2). A comparison of the two reactors is given in Table II-7. Both reactors employ hexagonal fuel assemblies, but there is a difference in the number of fuel pins per assembly. The movable fuel assemblies of the "compensating" type are for slow changes in reactivity. These consist of a flux-trap type of control rod, hexagonal in shape, mounted on top of a movable fuel assembly very similar in design to the stationary assemblies. The safety

States for a number of years. Other features of the WWER-2, notably the elimination of the cladding in the pressure vessel,²⁹ represent departures from practice in the United States.

The reference concludes with a discussion of future improvements in the reactor type not specifically tied to the WWER plant. One of the most significant areas for improvement is the method of refueling. Although WWER-1 had not been refueled at the time of the writing of Ref. 29, the authors were aware that, in general, refueling operations "...require a lot of time...." A number of possible solutions are discussed, and the most attractive one is a reactor design employing soluble poison since this application allows the removal of most, if

not all, of the control-rod-drive structure from above the core. The resulting reactor is shown in Fig. II-16. The "honeycomb" core design is not well explained in the reference, but the interesting application to refueling is the provision for storage of fuel elements within the pressure vessel. The fuel-element handling

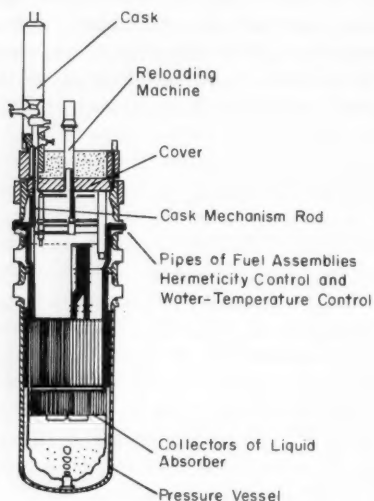


Fig. II-16 Reactor scheme with honeycomb core and with the change of liquid-absorber concentration in the channel system.²⁹

would be done after the reactor had been shut down but not depressurized. The centrally located reloading machine would handle the fuel elements, taking new elements from their racks within the vessel and replacing spent elements. After the fuel movement into and out of the core had been accomplished, the reactor would be brought to power, and the spent elements would be transferred through a vessel lock. The latter step need not be done rapidly since reactor operation would not be affected.

Several of the recommendations for advancement in the WWER technology are peculiar to the reactor as designed. In the WWER-1 the clearance between adjacent fuel bundles is only a few millimeters, and the requirement for the movement of the relatively large flux-trap control rods to a position above the core apparently makes the design of the upper grid plate somewhat difficult. If the fuel assemblies are locked in place at their lower end, the lack of open space within the core makes it difficult to

get at the locking devices with fuel-handling tools. Accordingly, Ref. 29 suggests that the fuel elements be hydraulically locked in place by a piston located at the bottom of the bundle but connected to the space above the core. The control rods would be bottom driven to facilitate refueling, and the reference illustrates a possible hydraulic-drive mechanism. The upward direction of the flow in the WWER reactor gives rise to what appear to be formidable hydraulics problems, other than the one of merely holding the fuel elements in place. The large fueled followers move against the coolant flow when they move in the direction of reducing reactivity, and they must be adequately cooled during any movement. Downflow is discussed in Ref. 29, but the designers appear reluctant to take the step due to problems connected with flow reversal during main pump coastdown.

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Section

III

Power Reactor Technology

Vessel Type Superheating Reactors

Status and Design Features

The status of the various nuclear superheat projects in the United States has been covered in several previous issues of *Power Reactor Technology*. In particular, Vol. 7, No. 3, presented the designs of the BONUS and Pathfinder integral boiling-superheating power reactors. Both these reactors are currently in the process of startup. Two experimental superheating reactors, BORAX-V and EVESR, have had some operating experience. The present state of the art of the U. S. superheat program is summarized in Ref. 1.

Table III-1 lists pertinent data for three integral boiling-superheating plants. The EVESR (Fig. III-1) is a separate superheat reactor in that it superheats steam from an outside source rather than evaporating and superheating within the same vessel. At present the saturated steam is being supplied by a fossil-fired boiler. The fuel is of annular geometry, as shown in Fig. III-2. Steam flows down the annular flow passage to the bottom plenum of the element, reverses flow direction, and flows up the central hole. Other details of the reactor are given in Table III-2.

Superheating reactors are being studied not only in the United States but also in Germany² and the Soviet Union.³ Reference 2 is a description of a prototype integral superheat reactor that was scheduled to be started in Germany in late 1964. The plant is to be 100 Mw(t) and is to produce steam with a temperature of 500°C at a pressure of 60 ata. The plant site is at Kahl, and the prototype reactor will power an existing turbine-generator set. The reactor core is composed of 45 fuel assemblies, arranged in a 7 by 7 lattice with the corners

missing (Fig. III-3). Each assembly is composed of 64 fuel elements of a hollow, annular design (Fig. III-4). The interesting coolant-flow path within the vessel starts with the liquid water undergoing heating and boiling on the exterior of the annular fuel elements, under natural-circulation flow. The saturated steam flows to the upper portion of the pressure vessel and enters the superheating passages of the center 25 fuel assemblies via appropriate piping. Each fuel assembly is two-pass in design, with steam flowing down the center hole of the outer 32 fuel elements and up the center hole of the remaining 32 elements. The partially superheated steam then flows to the outer 20 assemblies, shown shaded in Fig. III-3. The same two-pass flow within the assemblies takes place, producing steam of the final desired temperature. The entire reactor is thus essentially a four-pass superheater.

The combination boiling-superheating fuel element is an advanced concept and must be designed carefully with respect to thermal stresses. In the fuel element shown in Fig. III-4, a rather large temperature difference will exist between the walls in contact with the superheated steam and those in contact with the boiling water. In addition, the superheater wall temperature in the second-pass elements will be greater than the corresponding wall temperature in the first-pass elements. The latter problem is to be solved by bellows, and the other thermal stress is to be taken up by the use of proper material combinations to withstand the stresses. Elements of this general type have been constructed and irradiated in the United States; this program was reviewed in the March 1963 issue of *Power Reactor Technology*, 6(2): 75-80. Superheat tests have also been run in the Kahl boiling-water reactor and are briefly reviewed in Ref. 2. The

Kahl facility apparently is an in-pile loop capable of testing fuel elements, but not necessarily elements of full length. In February 1964, two fuel tubes were removed from the loop after a burnup of about 1150 Mwd/ton. These combination elements were clad with Inconel X and Hastelloy C, and both were "undamaged"; the maximum cladding temperature reached was relatively low—about 770°F. These tubes were fabricated from UO₂ pellets of annular geometry, although vibratory compaction is also being studied as a fabrication method.

Details of the Russian superheating reactor are given in Ref. 3. This plant is currently being completed in the Ulyanovsk region of the Soviet Union and will be an extremely flexible test facility. Two cores are presently contemplated for the facility—a small core (for

initial use) that will produce saturated steam at a pressure of 100 atm and a large core that will generate 90 atm of superheated steam at a temperature of about 500°C. The reactor as a whole is being designed to generate steam at pressures between 30 and 100 atm. Figure III-5 shows the plant flow diagram. The turbine is designed to operate with saturated steam at a pressure of 29 atm, and the high-pressure steam separator shown in Fig. III-5 provides for throttling of the steam, which will be generated at higher pressures. Superheated steam, when produced, can be delivered to a separate turbine or to a special cooler. The plant also provides for dual-cycle operation by the inclusion of steam generators in which 30 atm of saturated steam can be generated. Data on the reactor are given in Table III-3. Figure III-6 is a view of the reactor as arranged for super-

Table III-1 BOILING-WATER REACTOR PLANTS WITH INTEGRAL SUPERHEAT DESIGN DATA¹

	BONUS		BORAX-V		Pathfinder	
Site	Puerto Rico		NRTS, Idaho		Sioux Falls, S. Dak.	
Owner*	USAEC		USAEC		NSP	
Reactor designer*	GNEC		ANL		A-C	
Operating contractor*	PRWRA		ANL		NSP	
Electric power, Mw(e) gross	17.5		3.5		66	
Electric power, Mw(t) net	16.5		3.5		62.5	
Thermal output, Mw(t)	50		20		200	
Plant steam cycle	Direct		Direct		Direct	
Position of superheater region	Peripheral		Central or peripheral		Central	
Nominal operating pressure, psig	975		600		600	
Turbine throttle steam pressure, psig	850		350 (587 available)		535	
Saturated steam temp., °F	543		489		489	
Superheated steam temp., °F	900		850		825	
Thermal power to boiling, Mw	37		16.6		157	
Thermal power to superheating, Mw	13		3.4		43	
Gross cycle efficiency, %	35				33	
Net cycle efficiency, %	33				31	
Maximum superheat fuel-cladding temp., °F	1175		1110		1	
	<u>Boiling</u>	<u>Superheater</u>	<u>Boiling</u>	<u>Superheater</u>	<u>Boiling</u>	<u>Superheater</u>
Effective outside core dia., ft	3.34	4.83	3.25	1.0	5.75	2.5
Core height, ft	4.55	4.55	2.0	2.0	6.0	6.0
Structural material (core)	Zircaloy-2	S.S. 348	Al(X8001)	S.S.	Zircaloy-2	S.S.
Fuel type	Rod	Rod	Rod	Plate	Rod	Double tubular
Fuel material	UO ₂	UO ₂	UO ₂	UO ₂ -S.S. cermet	UO ₂	UO ₂ -S.S. cermet
Fuel enrichment, %	2.40	3.25	4.95	93	2.2	93
Cladding material	Zircaloy-2	Inconel	S.S. 304	S.S. 304L	Zircaloy-2	S.S. 316L
Control-rod shape	Cruciform	Slab	Cruciform and "T"	Cruciform and "T"	Cruciform	Round rod
Control-rod material	1.0 wt.% B ¹⁰ in S.S.	1.0 wt.% B ¹⁰ in S.S.	Boral	Boral	2 wt.% boron in S.S.	2 wt.% boron in S.S.
No. of steam passes		4		2		1
Average power density, kw(t)/liter	33.6	11.5	42.5	40.5	45.2	46.5

*GNEC = General Nuclear Engineering Corporation.

PRWRA = Puerto Rico Water Resources Authority.

ANL = Argonne National Laboratory.

NSP = Northern States Power.

A-C = Allis-Chalmers.

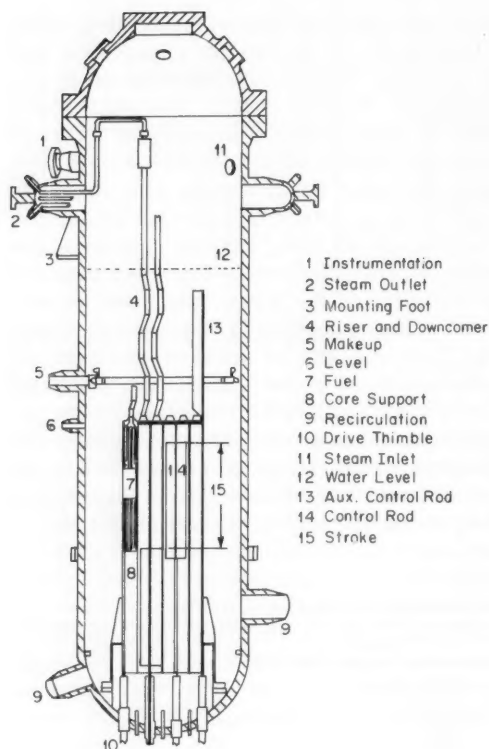
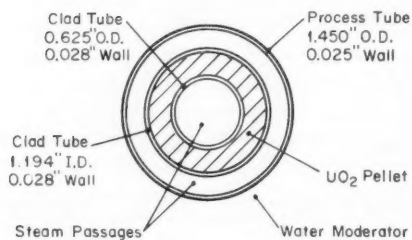
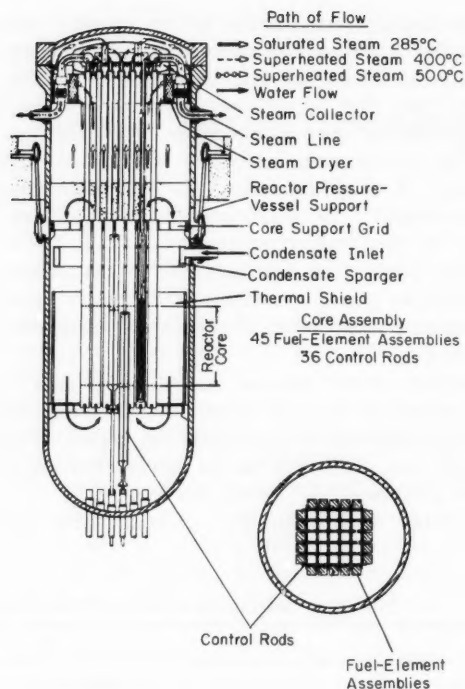
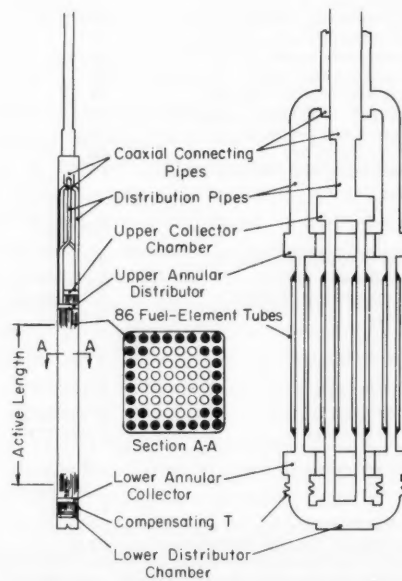
Fig. III-1 EVESR reactor.¹

Fig. III-2 EVESR fuel-element cross section.

Table III-2 DESIGN PARAMETERS OF THE EVESR REACTOR

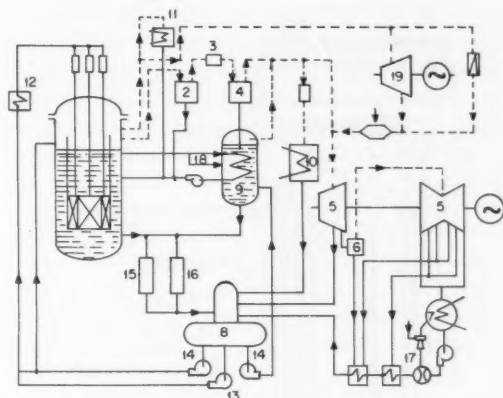
Type	Heterogeneous, light-water-moderated, thermal-neutron, steam-cooled
Power, Mw(t)	12.5 (initial)
No. of fuel elements	32
Process tubes/fuel element	9 (3 by 3 array)
Cladding	Incoloy, Inconel, S.S. 304 (commercial), S.S. 304VM, S.S. 310VM
Maximum cladding-surface temp., °F	1250

Fig. III-3 Schematic of the Kahl superheating reactor.²Fig. III-4 Schematic of the Kahl superheating-reactor fuel element.² ○, up pass. ●, down pass.

heating, and Fig. III-7 shows the cross section of the "large" core. The reactor is controlled by 31 movable assemblies for reactivity compensation and 6 safety assemblies. These appear to be similar in design and operation to those employed in the WWER, and their positions are illustrated in Fig. III-7. The follower employed in the safety assemblies is a zirconium scatterer, however, and is not fueled.

Table III-3 CHARACTERISTICS OF THE RUSSIAN INTEGRAL SUPERHEATING REACTOR

Pressure-vessel size		
Outside diameter, m	3.8	
Height, m	11.0	
Boiler fuel-assembly details		
Fuel	UO ₂	
Cladding	Zr alloy	
Cladding outside diameter, mm	10.2	
Cladding thickness, mm	0.6	
Fuel-element geometry	Hexagonal	
Distance across flats, mm	176	
No. of fuel pins/assembly	127	
Fuel-pin pitch, mm	15.1	
Superheater fuel-assembly details		
Fuel	UO ₂	
Cladding	S.S.	
Cladding outside diameter, mm	~10.2	
Cladding thickness, mm	0.5	
Fuel-element geometry	Hexagonal	
Distance across flats, mm	176	
No. of fuel pins/assembly	61	
Steam-flow-channel width, mm	1	
Insulating material	Stagnant steam	
	Large core	Small core
Power, Mw(t)	250	150
Core dimensions		
Diameter, m	2.6	1.8
Height, m	2.0	2.0
No. of fuel assemblies in the core		
Boiling zone	181	85
Superheater zone	109	
UO ₂ loading in the core, tons	72	
Boiling zone	19.3	11.4
Superheater zone	14.6	
Water-to-uranium ratio	4.7	
Boiling zone		
Boiling zone	1.5 to 2.0	2.0
Superheater zone	3.0 to 3.5	
Balance of reactivity, % k_{eff}		
Temperature effect (to 309°C)	2.4	3.5
Steam generation (for average steam fraction)	2.0	Up to 3.5
Doppler effect	1.6	2.0
Xenon and samarium poisoning (in equilibrium state)		
(in equilibrium state)	4.2	4.7
Fuel burnup	5.7	5.3
Structural elements	1.0	2.5
Total reactivity excess in a cold clean reactor	17.0	21.5
Worth of control devices in a cold reactor, % k_{eff} including:		
Absorbers with mechanical drives	19.0	24.0
Liquid poison	20.0	20.0
Maximum fuel-element heat flux, kcal/(m ²)(hr)		
	1.2×10^6	1.2×10^6



- 1 Reactor
- 2 High-Pressure Separator
- 3 Throttle Valve
- 4 Mean Pressure Separator
- 5 Saturated-Steam Turbine
- 6 Moisture Separator
- 7 Turbine Condenser
- 8 Deaerator
- 9 Steam Generator
- 10 Auxiliary Condenser
- 11 Shutdown Cooling Condenser
- 12 Heat Exchanger
- 13 Plunger Pump
- 14 Feedwater Pumps
- 15 Water-Purifier Evaporating Unit
- 16 Ion-Exchange Filter
- 17 Turbine Jet
- 18 Steam-Generator Feed Line with Step Evaporation
- 19 Superheated-Steam Turbine

Fig. III-5 Principal flow diagram of the Ulyanovsk plant.³

The two-pass superheating fuel element is illustrated in Ref. 3, but the print is of poor quality, at least in the copy seen by this reviewer. Figure III-8 approximates the fuel-pin design, however. The reactor is designed to utilize natural circulation. Construction apparently was well under way when downcomer entrainment was discovered to be a serious design consideration for boiling-water reactors, and according to Ref. 3 the downcomer entrainment will be 10 to 20%.

Consideration of Fig. III-6 indicates that the high downcomer velocity comes about primarily because of the placement of the reactor chimney around the entire core. The BONUS reactor, which is similar in a number of respects to the Russian design, has the core cage around the boiling zone only, and the downcomer velocities are low in the zone of steam-water separation.

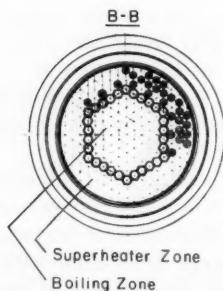
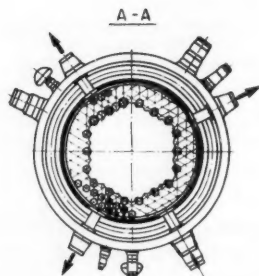
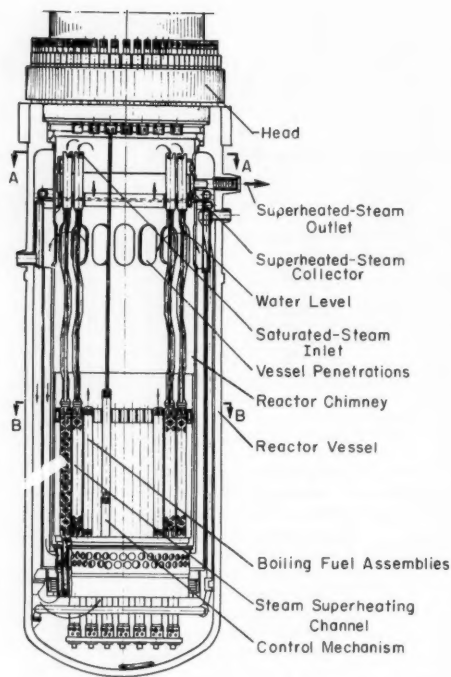


Fig. III-6 General view of the Ulyanovsk steam superheating reactor.³

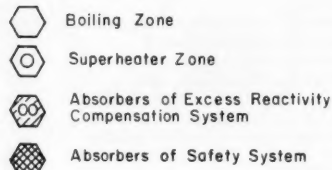
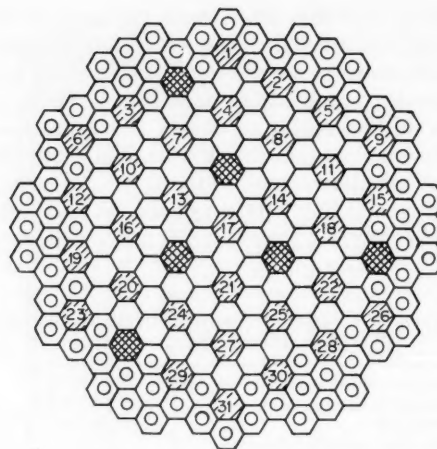


Fig. III-7 Fuel-assembly arrangement in the core of the Ulyanovsk superheating reactor.³

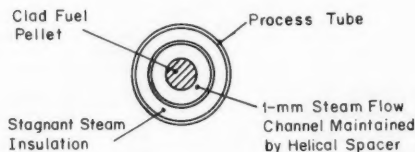


Fig. III-8 Cross section of superheater fuel element in the Ulyanovsk reactor.

Directions of Future Developments

Since no extensive operating experience has yet been gained with a pressure-vessel superheating reactor, it is difficult to comment on the performance of the concept as a whole, much less to predict future trends. In the United States a very considerable effort has been committed to the development of the reactor type, and both integral and separate designs have been constructed. The attention of the reader is directed to the superheat

write-up in the Summer 1964 issue of *Power Reactor Technology*, 7(3): 310-312, for a discussion of some of the pertinent problems pertaining to nuclear superheat with integral

reactors. With the exception of the nuclear coupling of the boiling-superheating portions of the core, the separate superheat concept exhibits qualitatively the same problems as the integral designs, although the steam-water separation problem may be somewhat different, and presumably more control can be exercised over the conditions of the steam fed to the superheating elements.

The combination boiling-superheating fuel used by the German designers has been discussed previously. A similar principle is being used for the second core of the BONUS reactor,⁴ and a typical fuel element is shown in Fig. III-9. The figure shows a design utilizing pelletized fuel, but vibratory compaction is also being considered.

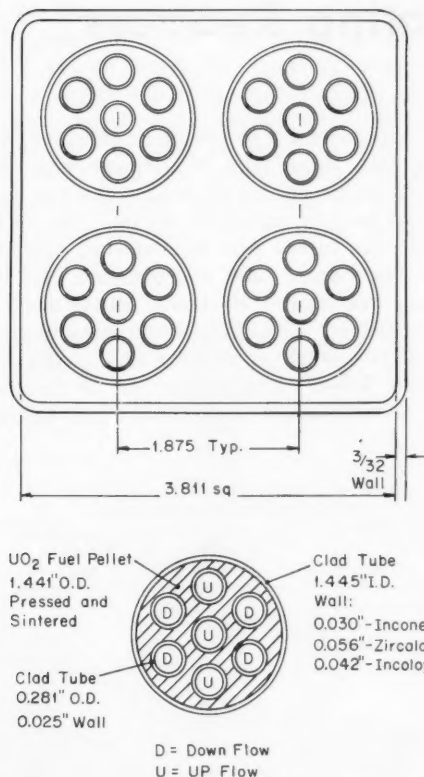


Fig. III-9 Typical fuel-assembly cross section for pelletized fuel in the second core of the BONUS reactor.⁴

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The references that contain A/Conf. numbers are papers that were presented at the Third United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, September 1964.

1. M. Novick, R. E. Rice, C. B. Graham, D. H. Imhoff, and J. M. West, Developments in Nuclear Superheat, A/Conf.28/P/215.
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3. I. N. Sokolov, V. I. Gritskov, P. N. Bogdanovich, E. V. Kulikov, S. A. Skvortsov, V. P. Spasskov, V. I. Plutinsky, I. A. Philateyev, and A. A. Hohla-chev, Experimental Pressure Vessel Reactor for Studies in Boiling and Steam Superheating, A/Conf.28/P/306.
4. J. F. Gibbons, Objectives and Mechanical Design Features of the Advanced BONUS Fuel, in Proceedings of the Ninth Nuclear Superheat Meeting, April 21, 22, and 23, 1964, San Jose, California, USAEC Report SAN-8003, San Francisco Operations Office, June 30, 1964.

Section IV

Power Reactor Technology

Graphite-Moderated Superheating Reactors

The existing reactor of the graphite-moderated superheating type is located at the Beloyarsk power station of the USSR. The reactor (called hereafter Bel-1) has been in operation since May 1964, generating electricity at reduced power. A second reactor of this type (called Bel-2) is currently being built at the same site. For the future outlook and the status of the graphite-moderated reactor to be understood, it is necessary to examine the early history of the Bel-1 plant.

The basic design of the Bel-1 type fuel element was established with the First Atomic Power Station (FAPS).¹ The fuel element for the FAPS reactor is shown in Fig. IV-1. In

FAPS the coolant was pressurized water, although this same type of element was also operated under boiling conditions and was used in a test loop to superheat steam.² The FAPS element was fueled with 5% enriched uranium alloy. The center tube was unfueled, and the uranium was canned as shown. All the tubes in the fuel channel were made of the Russian steel "1X18H9T," which is similar to the AISI 321 stainless steel.³ Dimensions of the various tubes are given in Table IV-1. The tubular

Table IV-1 DIMENSIONS OF THE FAPS
FUEL-ELEMENT COMPONENTS

Item	Diameter, mm	Thickness, mm
Central tube	15	0.6
Peripheral tube	9	0.4
Outer cladding	14	0.2

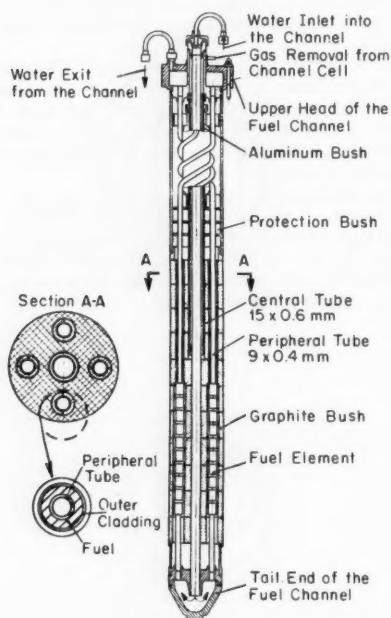


Fig. IV-1 Vertical section of the fuel channel in Russia's First Atomic Power Station.

spirals at the upper end of the FAPS element allow for relative movement between the central tube and the assembly of peripheral tubes. This would be particularly important if the element were used for superheating since the central tube wall would run at a temperature somewhat above saturation and the peripheral tube would operate at a temperature approximating the superheated-steam outlet temperature. The peripheral tubes, the fuel, and the outer cladding can operate as a unit; i.e., when the peripheral tubes expand, the fuel and the outer tube can move relative to the graphite surrounding the fuel tubes.

The fuel element used in Bel-1 is similar to the FAPS design (Fig. IV-1) except that the temperature-compensating coils are located at the bottom of the element.^{4,5} Reference 4 is the Second Geneva Conference paper on the design of Bel-1, and Ref. 5 is the Third Geneva Con-

ference paper on the development of both Bel-1 and Bel-2. In addition, the peripheral steam tube in Bel-1 was changed in size with respect to FAPS, so that it had a diameter of 9.4 mm and a thickness of 0.6 mm with an outer cladding diameter of 20 mm. The change in location of the expansion coils is not explained, and, in fact, they are placed back at the top of the element in the Bel-2 fuel-element design. The Bel-2 design will be discussed shortly. Instead of four peripherally located fuel elements, the Bel-1 fuel has six, all fed by a central tube. The same size and type of element is used for both boiling and superheating in Bel-1, but the materials may be different in the two applications.

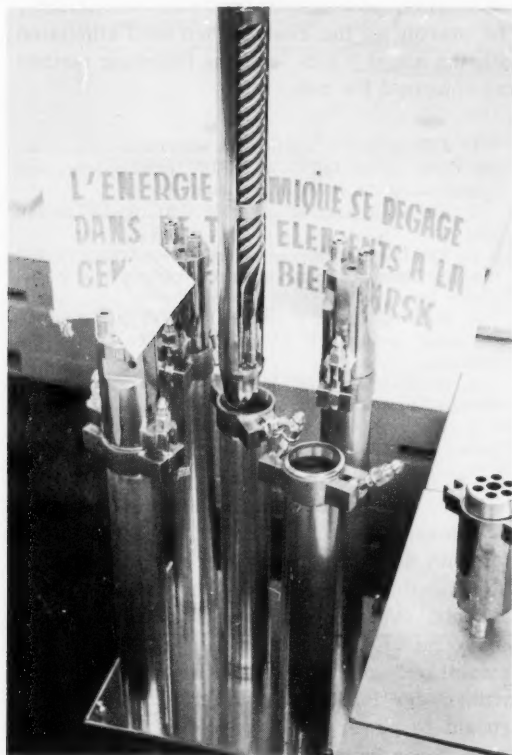


Fig. IV-2a The Bel-1 fuel element on display at the Russian Geneva exhibit. The photograph shows the lower end of a fuel element positioned over a process tube. The mockup shows adjacent process tubes, although for some reason they do not match, position-wise, the holes in the graphite logs. The light-colored blocks shown surrounding the assembly probably simulate the loading face of the reactor, although in the actual plant they are positioned above the location shown in the photograph.

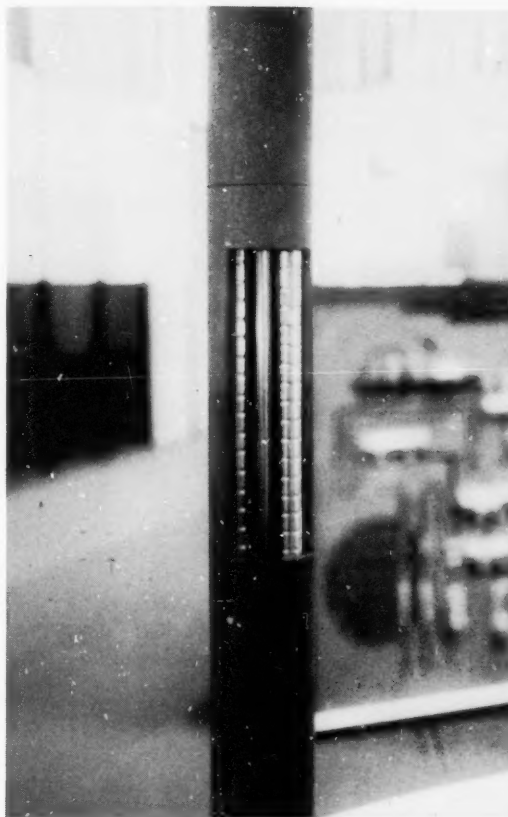


Fig. IV-2b The Bel-1 fuel element showing the upper portion of the compensating coils. The horizontal spacer shown is the same one located at the top of Fig. IV-2a.

Figures IV-2a, IV-2b, and IV-3 are photographs, taken by the reviewer, of the Bel-1 fuel element on display at the Russian exhibit in Geneva. The compensating coils are shown in Fig. IV-2b, and Fig. IV-3 is a cutaway of the fuel element in the uranium-bearing region. Figure IV-3 shows that the outer cladding is dimpled with circular indentations. Although the reasons for these indentations are not given in the references, it is presumed that they assist in giving axial support to the fuel and also act as bellows. There is a thermal-stress design problem arising from a difference in temperature between the inner and outer cladding tubes of the fuel annulus, which are welded together at their ends to contain fission gas. The inner cladding tube is steam cooled, whereas the outer cladding presumably is not. The



Fig. IV-3 The Bel-1 fuel element illustrating a cut-away view of the fueled region. Three peripheral superheating subelements are shown with the center one being sectioned to illustrate the fuel (dark, vertical portions) and internal cladding surface (shiny, vertical portion). The tubes at the left and right show circumferential indentations, probably for expansion.

fuel is a dispersion of uranium-molybdenum alloy in a magnesium matrix. The evaporating-channel characteristics for Bel-1 and FAPS are listed in Table IV-2, and Table IV-3 lists the superheating-channel characteristics.

An interesting feature of Bel-1 is that the steam to be superheated is not generated directly in the reactor but in a separate steam generator using the indirect-cycle principle. Details are shown in Fig. IV-4.

The permanent structure of the reactor consists of a graphite stack that has a diameter of 9.6 m and a height of 9.0 m. The basic building block is a 200- by 200-mm graphite log penetrated by an axial hole. The core contains 730 boiling assemblies, 268 superheating assem-

blies, 78 compensating control rods, 16 safety rods, 6 automatic control rods, 2 counting-chamber assemblies, 4 starting-chamber assemblies, and 30 ionization-chamber assemblies, for a total of 1134 core positions.⁴ Coolant conditions are given in Table IV-4.

Bel-1 achieved criticality in September 1963, and initial loadings were with boiler-only fuel both flooded and voided. A full core was loaded for critical experiments. This consisted of 500 evaporating elements enriched to 2%, 230 evaporating elements enriched to 1.5%, and 268 superheating elements enriched to 1.5%. According to Ref. 6, only 64 boron-steel compensating rods were used, however. A number of physics experiments were conducted with Bel-1, and these included power distributions, rod worths, power split, and void coefficients. The startup of the reactor had been simulated with the aid of FAPS,² and the following method was approved for use:

1. Filling the circuits with water and then heating them up to 180°C at 5–10% power level and establishing the level in the bubbler.
2. The plant heating up to 230°C at 10–15% power level.
3. Establishing the level in the evaporators and blowing through the superheating channels at 2–3% power level.
4. Raising reactor power level to 20% and increasing secondary circuit pressure up to the nominal.
5. Establishing the level in separators, i.e., bringing primary coolant to boiling.
6. Feeding steam to the turbine and further power rising.

To study this process with the Bel-1 reactor, the Russians removed all but two superheater elements and the startup procedure previously determined with FAPS was checked and found adequate. Voiding the superheating elements was found to take about 6 min and to proceed "smoothly." A total of 192 superheating elements were loaded, and the reactor was then brought to power to study voiding of this relatively large number. These results are indicated in Fig. IV-5. The "scavenging" operation mentioned in the figure apparently refers to the injection of a solution of hydrazine hydrate into the secondary circuit to control oxygen concentration. On Apr. 26, 1964, steam was fed to the turbine, but, since less than the normal number of superheating elements were still being used (192), the superheat temperature reached only 390°C (734°F). Future plans called

Table IV-2 EVAPORATING-CHANNEL CHARACTERISTICS⁵

	FAPS reactor	Bel-1 reactor	Bel-2 reactor				
			Zone I	Zone II	Zone III	Zone IV	Zone V
Channel power, kw	300	405	771	634	617	545	517
Coolant flow rate through the channel, kg/hr	2500	2400	5500	4700	4150	3550	3250
Steam void at channel outlet, %		33.6	27.6	29.3	30.5	32.1	34.2
Pressure at channel inlet, atm	100	155			155		
Pressure at channel outlet, atm	98	150			145		
Coolant temperature at channel inlet, °C	200	300			300		
Coolant temperature at channel outlet, °C	290	335			338		
Maximum thermal load, kcal/(m ²)(hr) ($\times 10^{-6}$)	1.8	0.5	0.8	0.7	0.6	0.5	0.5
Circulation rate, m/sec	4	3.5	4.6	4.0	3.5	3.0	2.7
Maximum temperature, °C							
Tube inner wall	324	355			365		
Fuel	382	400			415		
Ratio of burnout heat flux to maximum heat flux		2	1.85	1.9	1.9	2.0	1.95

Table IV-3 SUPERHEATING-CHANNEL CHARACTERISTICS⁵

	Bel-1 reactor	Bel-2 reactor
Maximum channel power, kw	368	767
Minimum channel power, kw	202	548
Steam flow rate through maximum power channel, kg/hr	1900	3600
Steam flow rate through minimum power channel, kg/hr	1040	2570
		Bel-2 downstream fuel elements*
Channel inlet pressure, atm	110	132
Channel outlet pressure, atm	100	125
Channel inlet steam temperature, °C	316	328
Channel outlet steam temperature, °C	510	399
Maximum thermal load, kcal/(m ²)(hr) ($\times 10^{-6}$)	0.48	0.83
Maximum steam velocity, m/sec	57	76
Maximum temperature, °C		
Tube inner wall	530	426
Fuel	550	482
Graphite	725	735
		Bel-2 upstream fuel elements*
		124
		110
		397
		508
		0.68
		112
		531
		565

* The terms "downstream" and "upstream" apparently should have been translated "downflow" and "upflow," respectively. Obviously the downstream element is the first-pass element, and the upstream element is the second-pass element.

for the addition of more superheat elements to increase the steam temperature to about 500°C.

The future plans for the reactor type introduce the reader to the second Beloyarsk reactor, Bel-2. The graphite stack for this reactor is similar to Bel-1 with respect to size and number of channels, but the second reactor should produce 200 Mw(e). This factor of 2 increase in power is startling at first considera-

tion, but its explanation is reasonably straightforward. The coolant cycle has been changed from indirect boiling to direct boiling, as shown in Fig. IV-6. Two possible arrangements are being considered and are shown as parts *a* and *b* in the figure. The difference between the two cycles is the presence (see part *a* of Fig. IV-6) of a bypass line that allows some of the feedwater to be injected into the steam sepa-

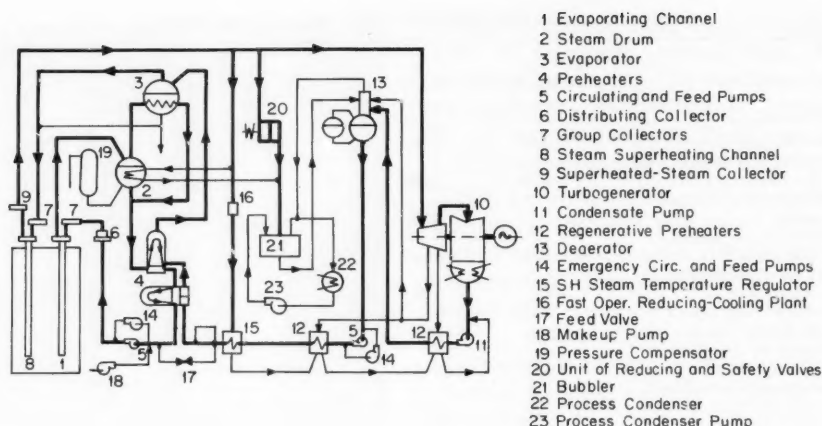


Fig. IV-4 Flow sheet of the first unit of the Beloyarsk Nuclear Power Station.⁶

rator after it has passed through the regenerative heater designated as the "steam-generator heater." This is said to be used for "steam washing," but its significance is not readily apparent. Besides the cycle change between Bel-1 and Bel-2, the elements of the latter are to be constructed with slightly larger coolant channels. These details are shown in Table IV-5.

Probably the most significant change between the two reactors is the superheater fuel for Bel-2. Just how the UO_2 is to be used in the Bel-2 superheater elements is not clear; the single mention of UO_2 in Ref. 5 is an offhand one, as follows:

Upstream superheating elements design and dioxide uranium-based fuel ensure steam superheating up to 500°C; this is confirmed by loop-tests on the first power plant reactor.

On the basis of Table IV-3, we assume that the intent is to substitute UO_2 for the uranium-molybdenum alloy in the dispersion type element, for the maximum fuel temperature listed is far lower than that which would characterize a solid oxide fuel body. It is disappointing that the information on the details of the Beloyarsk elements is so limited, for it is this aspect of the reactor which is probably of greatest interest to designers outside the USSR.

The mechanical design of the Bel-2 superheater elements has also been changed from that of Bel-1. The central, unfueled tube feeding steam to the peripheral channels in the Bel-1 element has been removed and the ele-

ment converted to two-pass design. These details are shown in Fig. IV-7. As a result the steam velocity is greatly increased, from 57 to 112 m/sec (Table IV-3), and the improved heat-transfer coefficient allows an increase in power. In place of the central tube, an absorber pin is included in the Bel-2 element. These absorbers, which probably are manually manipulated, make possible a reduction of the radial power-peaking factor for the core to 1.3, compared to a value of 1.4 in Bel-1 (see Ref. 5). The evaporating elements for Bel-2 retain the center tube for coolant flow. The increase in power from the boiling portion of the core appears to be the result of rather careful zoning

Table IV-4 MAIN CHARACTERISTICS OF THE REACTOR COOLANT IN THE BELOYARSK NUCLEAR POWER STATION

Characteristics		
Reactor thermal output, Mw	286	
Electrical output, Mw	100	
Ratio of superheating- to evaporating-channel output, %	30.0	
	First circuit	Second circuit
Coolant flow rate, tons/hr	1200	405
Coolant pressure, atm		
At reactor inlet	155	110
At reactor outlet	150	95
Before the turbine	150	90
Coolant temperature, °C		
At reactor inlet	300	316
At reactor outlet	340	510
Before the turbine	340	500
Average steam quality at channel outlet, wt. %	33.6	

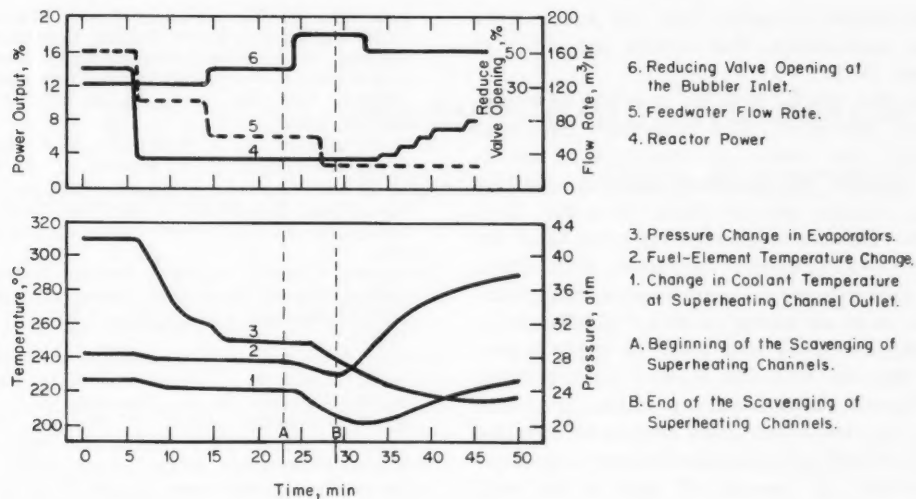


Fig. IV-5 Curves of changes in the main parameters of the plant during transition to steam superheating conditions.⁶

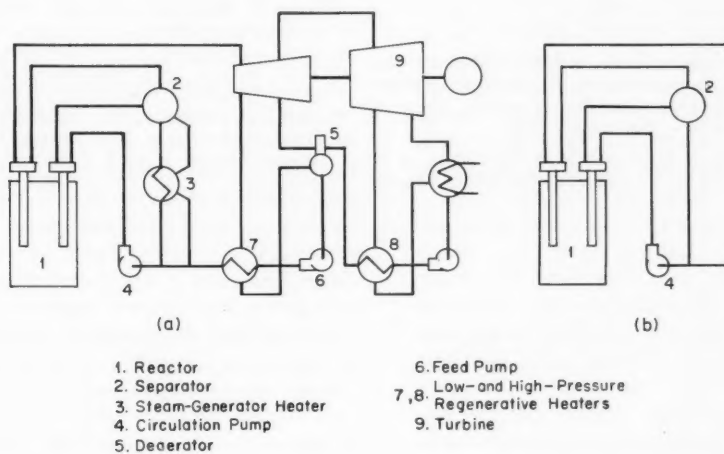


Fig. IV-6 Possible flow sheets for uranium-graphite tube type nuclear superheat reactors.⁵

Table IV-5 COMPARISON OF BEL-1 AND BEL-2 BOILING AND SUPERHEATING FUEL ELEMENTS

	Bel-1	Bel-2
Boiler and superheater fuel-element peripheral-tube diameter/thickness, mm	9.4/0.6	12/0.6
Fuel material		
Boiler element	U-Mo-Mg	U-Mo-Mg
Superheater element	U-Mo-Mg	UC ₂

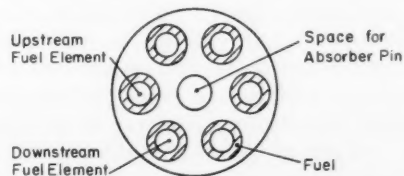


Fig. IV-7 U-shaped superheating channel cross section.⁵

of the reactor to match flow and power of the boiling assemblies. The results are indicated in Table IV-2.

The step beyond Bel-2 is stated to be supercritical operation. The tube type Beloyarsk reactors are suited to this type of operation because the pressure-bearing portions of the reactor are of small diameter. Even so, the cladding begins to get quite thick for supercritical-pressure operation, with attendant thermal-stress problems during transients. It is estimated in Ref. 5 that the Bel-1 and Bel-2 reactor size could be made to produce 800 to 1000 Mw(e) if a supercritical, once-through design were possible. It is also suggested that an advanced reactor be operated with partially unclad fuel to remove fission gas and reduce the amount of steel in the core. This would be accomplished by removing the outer cladding tube, although relative movement of the coolant tube with respect to the graphite would have to be provided for.

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The references that contain A/Conf. numbers are papers that were presented at the Third United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, September 1964.

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Section

V

Power Reactor Technology

Heavy-Water-Moderated Pressure-Tube Reactors

The effort on heavy-water-moderated pressure-tube reactors throughout the world is indicated in Table V-1. The Canadian effort is by far the largest and the farthest advanced. The two Canadian power reactors, NPD and CANDU, are both cooled and moderated by D_2O , are fueled with natural UO_2 clad with Zircaloy-2, and employ calandria tanks to contain the moderator. The calandria tubes of NPD are aluminum, whereas those of CANDU are nickel-free Zircaloy-2. Design features of NPD are given in Ref. 1, and those of CANDU, in Refs. 2 and 3.

Until the recent announcement⁴ of a program directed toward the development of a 300-Mw(e) organic-cooled plant, there had been no definite plans for a large-scale heavy-water reactor in the United States. Nevertheless, the U. S. effort has been considerable and has resulted in the construction of a 17-Mw(e) prototype (the Carolinas-Virginia plant, CVTR), as well as a reactor that operates at comparable temperatures and thermal-power levels for plutonium-recycle development (the Plutonium Recycle Test Reactor) and a reactor for developing fuel elements for D_2O power reactors (the Heavy Water Components Test Reactor). All these reactors are cooled and moderated by heavy water. The Carolinas-Virginia reactor was described in *Power Reactor Technology*, 6(4): 63-81; the most recent general review of heavy-water reactors was in 7(1): 85-98.

Interest in heavy-water-moderated reactors is also high in other countries, but there is considerable diversity in the choice of coolant. The Steam Generating Heavy Water Reactor (SGHWR)⁵⁻⁸ utilizes boiling light water, the DON reactor uses organic coolant,⁹⁻¹¹ whereas EL-4 (Refs. 12 to 15) and the Lucens reactors¹⁶⁻¹⁸ are cooled with CO_2 . Of the reactors under study or development, the DOR,^{19,20} the R-1 and R-2, (Ref. 21), and the U. S. project²² are all designed to be organic cooled. The

German reactor study involved gas cooling.²³ In addition to these power reactors, at least two test reactors are being constructed to study the problems arising from organic cooling. These are the Canadian WR-1, reviewed in the Summer 1964 issue of *Power Reactor Technology*, 7(3): 324-327, and the ESSOR.²⁴

Operating Experience

Extensive operational experience has been reported^{25,26} for the NPD reactor. Reference 25 contains information on reactor physics aspects of the startup. Various reactivity coefficients were determined, including the moderator-level coefficient of reactivity, reflector reactivity worths, and temperature coefficients of reactivity. The reactivity balances at two different points in the lifetime are shown in Table V-2.

As is indicated in Table V-2, the sources of reactivity adjustment, other than fuel management, are quite limited, consisting only of changes in moderator level and moderator temperature. The normal NPD fueling operation consists of inserting a bundle in one end of a pressure tube, thereby shifting all the bundles within the tube by one bundle length and expelling the last bundle. This bundle may then be either discharged or recycled in a channel fueled in the opposite direction. This operation was studied by means of a computer code simulating the reactor, STOKE, which calculated the burnup of all fuel bundles by increments, taking fuel shifts into account. The code was designed to select the best shift for the current state of the reactor, consistent with the particular refueling scheme being studied. The original NPD loading consisted of natural and depleted UO_2 . The fuel-management principle consisted of shifting fuel in the channel containing the bundle with the highest burnup and recycling an expelled bundle

Table V-1 D₂O-MODERATED PRESSURE-TUBE REACTORS

Reactor	Country	Status*	Mw(e)	Coolant	Fuel and geometry	Cladding	Core type	Pressure tube	Core orientation	No. of refueling machines	Refueling method	Pressure-tube joints
NPD	Canada	Operational	19.5	D ₂ O (U)	UO ₂ rod cluster	Zircaloy-2	Calandria	Zircaloy-2	Horizontal	2	On-power	Expanded
CANDU	Canada	Being built (1965)	203	D ₂ O (U)	UO ₂ rod cluster	Zircaloy-2	Calandria	Zircaloy-2	Horizontal	2	On-power	Expanded
CVTR	U. S.	Operational	17	D ₂ O (U)	UO ₂ rod cluster	Zircaloy-4	Cold U tubes	Zircaloy-4	Vertical	1	Shutdown, depressurized	Conoseal gasket
SGHWR	England	Being built (1967)	100.0	H ₂ O (boil)	UO ₂ rod cluster	Zircaloy-2 (boiler), S.S. (superheater)	Calandria	Zircaloy-2	Vertical	1	On-power	Expanded
EL-4	France	Being built (1967)	70	CO ₂	UO ₂	S.S. or Be	Cold tube	Zircaloy-2	Horizontal	2	On-power	Threaded and brazed
DON	Spain	Planned	30	Organic	UC, rod cluster	Finned SAP	Calandria	SAP	Vertical	1	Shutdown, depressurized	Hot rolled and/or diffusion bonded
Lucens	Switzerland	Being built (1966)	13	CO ₂	U-1% Cr	Mg-0.5% Zr	Calandria	Zircaloy-2	Vertical	1 plus a locking machine	Shutdown, depressurized	
DOR	Denmark	Study	235	Organic	UO ₂ or UC	SAP	Calandria	SAP	Vertical	Probably 1		
R-1	Russia	Study	500	Organic	Beta-U	Mg-Be	Calandria	SAP	Vertical	Probably 1		
R-2	Russia	Study	500	Organic	UO ₂	SAP	Calandria	SAP	Vertical	1		
OCHWMR	U. S.	Study	512	Organic	UO ₂	SAP	Cold tube	Zircaloy-2	Vertical	1		
AKB	Germany	Study	100	CO ₂	UO ₂ or UC	Steel, Zr or Be alloys	Cold tube	Zr-Cu-Mo-lined	Vertical	1		

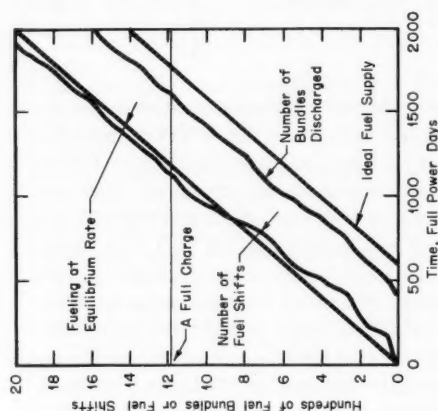
*The year in parentheses is the probable completion date.

Table V-2 CHANGES IN REACTIVITY BALANCE WITH IRRADIATION²⁵

	Reactivity effect in unirradiated core, mk	Reactivity effect in irradiated core (230 full-power days), mk
Leakage at full tank	-71	-69
Control margin*	-2.8	-0.3
Depleted fuel†	-23	0.0
H ₂ O in D ₂ O (relative to 100% D ₂ O)	-10	-27.0
Xenon	-23.9	-24.1
Samarium	0.0	-8.1
Moderator temp. (21-60°C)	-2.8	+0.4
Coolant temp. (21-254°C)	-7.4	-2.6
Power (fuel temp.)	-3.1	-1.4
Total reactivity load	144	132
Fuel burnup		12

*Operating moderator level is below full tank by amounts shown.

†Depleted fuel removed prematurely.

Fig. V-1 Simulation of NPD fueling, assuming that the fuel is recycled if the exit burnup is less than half the equilibrium burnup.²⁵

if it contained natural uranium and had a burn-up less than a specified fraction of the burn-up of the discharged fuel at equilibrium. The simulation of the NPD fueling, assuming recycle if the exit burnup was less than half the equilibrium burnup, resulted in the data shown in Fig. V-1. This figure predicts the number of fuel shifts and number of bundles discharged (and fed), assuming the NPD starts from its initial arrangement of partially depleted fuel. In practice, however, the NPD ran with depleted uranium fuel from its initial criticality in April 1962 to December. At that time the moderator and coolant became contaminated with light water, as will shortly be discussed, and the depleted fuel was removed to compensate for the resulting reactivity loss.

The degrading of the D_2O was caused by a major leak at the seal between the fueling machine and the pressure tube the first time on-power fueling was attempted. The spilled heavy water was downgraded from 99.7 to 99.1% D_2O and was contaminated by oil and grout. An upgrading program was initiated, and the purity was about 99.6% as of the writing of Ref. 26. The reference also states that loss and downgrading of the moderator have been negligible and that practically all D_2O losses and downgrading have come from the heat-transport system. The following quotation summarizes the D_2O loss experience on NPD:²⁶

- (a) Low escape rates depend a great deal on quality of the original design and construction.
- (b) Quite high chronic escape rates are acceptable if recovery is efficient i.e. if the "dry-room principle" is followed.
- (c) At a large station, it is not economical to sacrifice capacity factor in order to repair leaks. Consequently, if leaks are bad, recovery is essential.
- (d) There is no valid experience to show that escape rates can be held to extremely low values over long periods of time, without serious loss of capacity factor.
- (e) With efficient recovery, loss rates are virtually independent of escape rates, and can be very low.
- (f) NPD experience has shown that use of vapour-recovery equipment is both feasible and sensible.
- (g) With efficient recovery, the loss rate can be virtually unaffected by station size.
- (h) The "dry-room principle" not only is valuable for the everyday situation, but would have reduced the cost of the acute escape on 3 Dec. 1962 by an order of magnitude.
- (i) D_2O losses can be maintained acceptably low in any station whose design is based on NPD experience, and D_2O escape rates can be significantly reduced.

Fuel and pressure-tube performance have been good. Other than the major failure that occurred during the first refueling attempt—the refueling machines have been subsequently modified to prevent a future occurrence—the fueling-machine experience has been "highly successful."

Significant Features

Fuel and Cladding Materials

In support of NPD, CANDU, and various heavy-water-moderated power-reactor studies, the Canadians presented several important papers on the behavior of fuel and cladding materials.²⁷⁻³⁰ Some of the details pertain explicitly to NPD and CANDU type reactors, but others are of general interest to reactor designers. A rather interesting conclusion²⁷ was that high UO_2 density may not be an important goal for operating fuel elements, since low-density material (less than 10.3 g/cm³) provides for the retrapping of escaped fission gas. The elimination of the need for a plenum or other space within a fuel element to accommodate fission gases may be an important design consideration for certain fuel elements—particularly for the typical short Canadian elements, since plenums at the ends of the elements tend to cause power peaks in the fuel adjacent to the plenums. Reference 28 details the performance of the NPD and CANDU fuel elements and contains some interesting cost data. On the basis of production experience with the NPD and CANDU fuel elements, it is estimated²⁸ that the average cost of fuel for a 1000-Mw(e) CANDU type reactor would be \$50 per kilogram of uranium $\pm 10\%$. This results in a fuel cost of about 0.8 mill/kw-hr if the fuel is discarded without reprocessing.

Reference 30 discusses in-pile results for test fuel elements cooled by organics. Test elements were manufactured with sintered-aluminum-product (SAP) cladding containing 7 and 10% oxide (SAP 930 and 895, respectively) and a material apparently from another producer also containing 7% oxide, M 257 or M 583. The rods were designed to be freestanding under coolant pressure (300 psi) and with sufficient diametral fuel-jacket clearance to theoretically produce no diametral sheath strain at operating conditions. This was done because of the inability of SAP to deform plastically. The results of the irradiations are given in Table V-3.

Table V-3 RESULTS OF IRRADIATIONS IN TERPHENYL MIXTURE IN NRX TEST LOOP³⁰

	Test 704-I	Test 704-II	Test 709-I	Test 709-II	Test 712	Test 713	Test 714-I	Test 714-II	Test 719	Test 720
Sheath material	M 583	M 583	M 583	M 583	M 257	M 257	SAP 930	SAP 930	Zircaloy-2, Zr-2½% Nb	SAP 895
Element length, cm	30.5	30.5	61	61	122	122	244	244	50	22
Diametral clearance, cm	0.009 to 0.019	0.019	0.013	0.013	0.015 to 0.018	0.015	0.016	0.016	0.008 and 0.015	0.0025, 0.0050, 0.0075, and 0.0125
Irradiation, full-power days	<1	31.5	63.7	<1	59.3	29	44.6	45.6	82	4
Maximum $[k d \theta]$, watts/cm	~38	45.5	37.8	45	36.5	36.5	37.5*	39.9	39	40
Average $[k d \theta]$ at flux center, watts/cm	39	39	35	420	33.4	34	33.9	38.5	35	40
Average sheath temp., °C	420†	~430†	450	420	470	440	470	470	475	462
Maximum sheath temp., °C	†	†	465	470	496	484	497	525	500†	462†
Maximum diameter change, %	§	§	0.3	0.8	0.4	†	1.5	0.5	1.2	0
Maximum diameter change, %	§	§	~0	0.3	<0.2	†	0.5	~0	0.7	0
Average diameter change, %	§	§	0.03	0.05	0.04	†	†	~0	0.3	0
Average length change, %	§	§	2400	2100	1000	1500	1500	1500	2600	130
Burnup at flux center, Mwd/metric ton	1100	1100	2400	2100	1000	1500	1500	1500	2600	130
Fuel-sheath heat-transfer coefficient, watts/(cm)(°C)	†	†	~0.4	†	~0.4	†	~0.5	†	†	†
Fission-gas release, %	†	†	~3	†	~9	†	†	†	†	†
Fouling-film thickness, μm	10 to 500	10 to 1400	40 to 80	<5	<1	†	<1	<1	<1	<1
Ridge height, μm	**	**	**	**	**	**	40 max.	40 max.	30 av., 60 max.	0
Performance	Failed††	Good	Good	Failed††	Good	Good	Failed§§	Failed§§	Good	Good††

*A transient peak in excess of 40 watts/cm may have existed at the time of failure.

†No sheath thermocouples; temperatures listed are from calculation.

‡Estimated.

§Elements were stored under water and corroded before measurements were made.

*†Not measured.

††Circumferential ridges were observed but not measured in these tests.

‡†Failure was due to burnout. One element did not have ribs or wire wrap and bowed out and touched the pressure tube. This error was corrected in 704-II.

§§709-II contained poor-quality sheathing rejected for use in 709-I. This and the gross overpower are suspected to have led to failure.

**Failure appeared as stress cracks at the axial center of the elements. The cause of these stress cracks is unknown.

††Diametral-clearance experiment.

Although the causes of the stress cracks (tests 714-I and 714-II) were unknown, the reference notes that sheath strain probably did exist, due to migration of cracks within the fuel until the diametral clearance was eventually transferred to a central void. Evidence of a reaction between UO_2 and SAP was observed, and this was also noted in Ref. 31. The good performance of the zirconium-niobium clad UO_2 , illustrated in Table V-3, shows promise for the material, but further experiments were planned to test the hydriding behavior.

the exception of that shown for EL-3. Soluble poisons dissolved in the D_2O were studied, namely boric acid and the sulfates of cadmium, lithium, and gadolinium. Although the concept appeared feasible, the use of a so-called "gaseous" control rod, composed of ^3He , is discussed for application to EL-4.

Canadian experience with the decomposition of D_2O in various reactors is summarized in Table V-5. The NRU and NPD results shown in the table illustrate that increasing acidity and temperature both decrease the equilibrium con-

Table V-4 COMPOSITION OF MAGNESIUM-BASED ALLOYS SUITABLE FOR CLADDING OF GAS-COOLED REACTORS³³

Type of alloy	Composition of alloys, wt.% (balance Mg)									
	Be	Si	Al	Th	Fe	Mn	Ni	Cu	MgO	Ca Zr
PMB	From 0.5 to 32		0.01		0.04		0.001	0.005	0.2	
MB-3	0.04	0.5			0.01	0.001	0.001	0.005		
MB-4	0.08 to 0.15	0.7			0.01	0.001	0.001	0.005		
Mg-Al-Be	0.04		0.5		0.01	0.001	0.001	0.005		
Mg-Th-Be	0.04			3	0.01	0.001	0.001	0.005		
Mg-Ca-Zr-Be	0.04				0.01	0.001	0.001	0.005		0.5 0.5

References 32 through 34 are concerned with metallurgical problems of other cladding materials. The FeAl40, which is an iron-aluminum alloy containing 40 at.% aluminum, is a possible cladding for EL-4, the French gas-cooled reactor. The material offers possibilities as a substitute for beryllium, although it apparently has low ductility.³² The reference gives mechanical-property data and discusses welding and corrosion resistance. The magnesium-beryllium alloy described in Ref. 33 probably is the alloy mentioned for the Russian organic-cooled reactor R-1 (Ref. 21), although in Ref. 33 it is presented as a possible cladding for a gas-cooled reactor. Several alloys were considered (Table V-4), in composition ranges similar to the British Magnox alloys.

Water Chemistry—Experience with D_2O

A key consideration in the heavy-water-moderated reactor is the technology of the moderator itself, and three papers were devoted to this subject.³⁵⁻³⁷ Figure V-2 summarizes French experience with a number of low-temperature D_2O -moderated reactors with respect to moderator degradation. The figure apparently pertains to the situation in which no on-stream upgrading was attempted, with

concentrations of dissolved radiolytic gas. The problem of maintaining a nonexplosive atmosphere in spaces adjacent to the moderator water is not a trivial one and must be considered during reactor design. The NPD reactor moderator chemistry is also complicated by irradiation-formation of nitric acid from nitrogen introduced in air leakage.

Canadian experience in upgrading D_2O is contained in Ref. 37. The various Canadian reactors contain about 200 metric tons of D_2O altogether, and a number of incidents over the years have resulted in the reprocessing of many metric tons of degraded D_2O . The various methods discussed include electrolysis and distillation; distillation is the favored method for on-stream upgrading of reactor moderator because of its flexibility. A significant hazard in handling irradiated heavy water is tritium, the product of neutron capture by deuterium. In the NPD reactor the tritium concentration has reached 1.7 curies/liter and is expected to reach at least 10 curies/liter within a few years. Although the problem can be controlled by design and administrative procedures, it is one which is peculiar to the reactor type. A review of the economics of heavy-water production in the United States is given in Ref. 38.

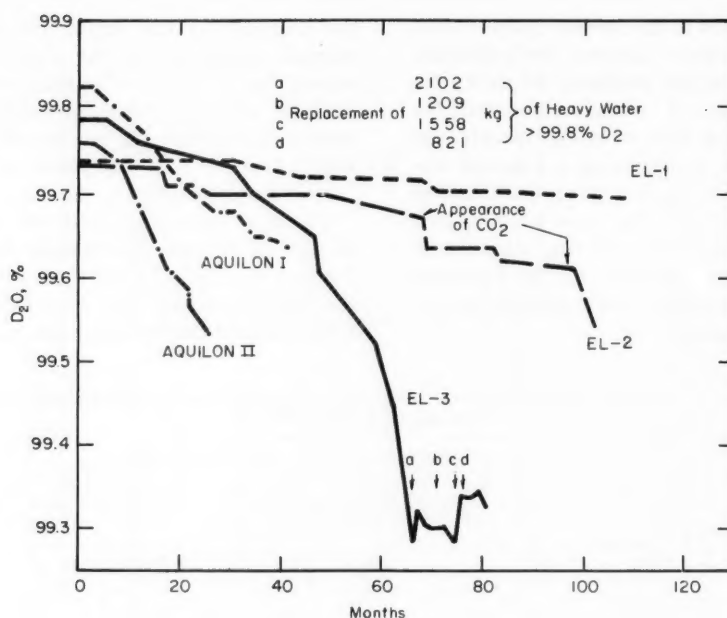


Fig. V-2 Evolution of the isotope content of heavy water in the C.E.A. reactors.³⁵

Table V-5 RADIOLYTIC DECOMPOSITION OF HEAVY-WATER MODERATORS³⁶

Reactor	Moderator temp., °C	pD*	D ₂ content of D ₂ O in core, cm ³ /kg	Total gas to recombiner, liters (STP)/min	Water decomposition	
					Liters of D ₂ (STP)/min	Grams of D ₂ O/Mw-hr†
NRX	45	~7	1.6	370	0.15	0.2
NRU						
1960-61	50	~7	5	230	6	1.6
1962-63	50	~7	2	230	3	0.8
1963	50	5.7	1	200	1.6	0.4
NPD	80	~6	1.2	280	8.5	5.1
	40	5.5	3.3	700	27	16

*In NRX and NRU pD of 7 corresponds to a conductivity of 0.2 megohm/cm (25°C); acidic conditions were due to DNO₃ added to NRU for a two-month experiment and formed in NPD from air in the helium.

†Megawatt-hours of total reactor thermal power; average power densities are 2.6, 5.0, and 1.7 watts/g for NRX, NRU, and NPD, respectively.

Pressure Tubes

Most of the reactors listed in Table V-1 utilize Zircaloy-2 as the pressure-tube material. Two papers were presented dealing with Zircaloy-2 as applied to pressure-tube construction,^{39,40} and two papers were concerned with the zirconium-niobium alloy.^{41,42}

The pressure tubes in NPD are seamless Zircaloy-2 tubing, fabricated by hot extrusion and cold drawing.³⁹ Dimensional gauging and borescope examinations were performed on one of the tubes after a year's service in the reac-

tor, and no significant changes were noted. The NPD design stress was chosen as 9.5 kg/mm² at 280°C, whereas the CANDU design stress was selected as 11.2 kg/mm² at 300°C. Tensile tests on portions of the NPD pressure-tube materials have given the results shown in Table V-6. In the reactor, however, the tubes are biaxially stressed by the combination of transverse and longitudinal loads produced by the pressurized coolant, with the result that the burst strength of the tube is considerably higher than would be predicted from simple tensile-test data; this is an important design consideration.

Table V-6 TENSILE-TEST RESULTS—NPD PRESSURE-TUBE MATERIAL³⁹

(Temperature = 280°C)		
	Longitudinal properties	Transverse properties
Ultimate strength, kg/mm ²	37.3	35
0.2% yield strength, kg/mm ²	31.6	33
Elongation, %	22.5	21
Reduction in area, %	45	60

The effects of hydrogen pickup by Zircaloy are complicated; they are also important, since the operating lifetime of a pressure tube is expected to be determined by the amount of hydrogen pickup experienced during corrosion. Hydrogen concentrations up to 250 ppm had little effect on mechanical properties of irradiated and unirradiated Zircaloy-2 so long as the hydrides were randomly arranged. These data are shown in Table V-7. It was established that prior strain, fabrication history, and stress were important in determining hydride orientation.³⁹

... It was found that hydrides precipitated while Zircaloy was under stress, aligned themselves perpendicular to tensile stresses and parallel to compressive stresses and that directional hydrides significantly decreased the mechanical properties of Zircaloy-2 ...

The use of zirconium-niobium alloys as pressure tubes is receiving study in both Canada⁴¹ and the United Kingdom.⁴² The advantage of the niobium alloy is in increased strength; properly heat-treated zirconium-2.5

wt.% niobium attains a strength about 50% higher than Zircaloy-2. These data are illustrated in Fig. V-3. The corrosion and hydriding resistance of zirconium-niobium alloys and Zircaloy-2 are similar, as indicated in Fig. V-4, although the impact strength of the niobium-bearing alloy is higher than that of Zircaloy-2. It appears that the zirconium-niobium alloy may become the preferred material for pressure tubes in cases where zirconium-base alloys are applicable.

The ternary alloy zirconium-2.5 wt.% niobium-0.5 wt.% curium is being developed because of its good corrosion resistance in moist CO₂ and is used in CANDU as a pressure-tube spacer. This spacer maintains clearance between the pressure tube and the calandria, and in CANDU this space is filled with the moist CO₂.

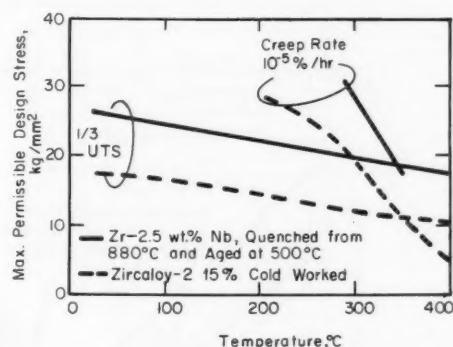
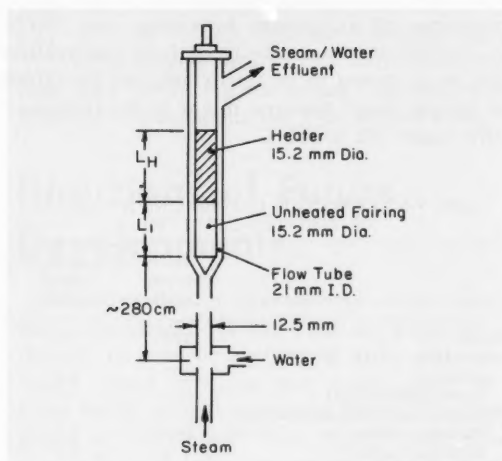
Fig. V-3 Design stresses for zirconium alloys.⁴¹

Table V-7 EFFECT OF NEUTRON IRRADIATION AND HYDROGEN CONCENTRATION ON THE PROPERTIES OF ZIRCALOY-2 (REF. 39)

(Each Result Is An Average for Three Specimens)

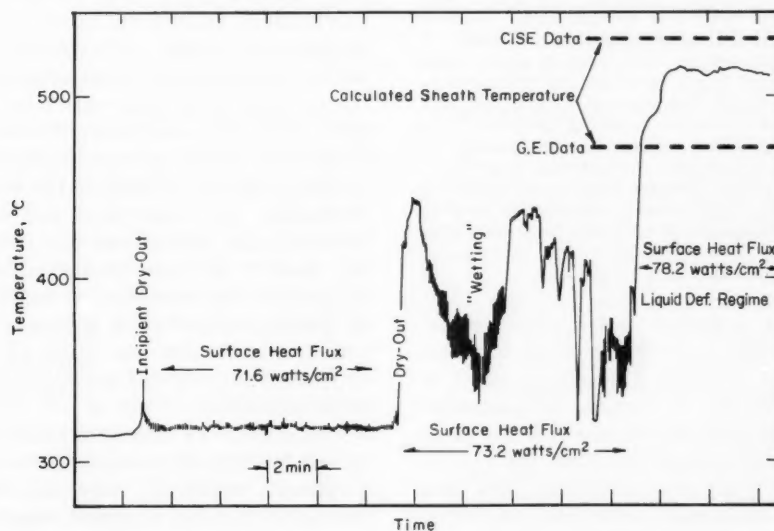
Testing temperature	Hydrogen content, ppm	0.2% proof stress		Ultimate tensile strength		Elongation	
		Unirradiated, kg/mm ²	Irradiated, kg/mm ²	Unirradiated, kg/mm ²	Irradiated, kg/mm ²	Unirradiated, %	Irradiated, %
Room temperature	0	56.6	77.3	66.1	78.7	12.4	5.3
	20	62.0	85.8	70.0	85.8	18.5	6.8
	100	61.8	83.4	70.3	83.7	15.2	7.3
	250	60.5	84.4	70.7	84.4	14.1	6.2
300°C	0	29.7	45.0	33.5	45.0	12.8	5.8
	20	30.1	47.1	33.1	47.1	17.9	7.4
	100	29.7	45.7	33.60	45.7	21.8	8.5
	250	32.2	46.4	36.34	46.4	19.0	8.8
400°C	0	27.5	37.3	29.0	37.3	11.1	6.8
	20	26.5	37.3	28.1	37.3	8.7	10.5
	100	25.9	36.6	27.5	36.6	18.0	9.9
	250	27.2	36.6	28.4	36.6	17.4	10.7

Fig. V-5 Test section.⁴³

are noted (incipient dry-out). The power is then adjusted to allow a small temperature oscillation (~ 5 deg C) and a set of readings taken. At the end of this period a slight increase in the test-element power ($\sim 2\%$) resulted in a sharp increase in surface temperature (~ 100 deg C)—the dry-out point. Continued operation here gave very large temperature oscillations, despite no recorded changes in the steady-state value of the heat flux, quality or flow. It would appear that there is an alternate wetting and drying out of the surface. A further increase in power ($\sim 7\%$) gives a further rise in surface temperature (up to 512°C). However, in this case there are no oscillations and the surface now appears to be "dry." ...

Figure V-7 illustrates the effect of the unheated fairing length, L_1 (see Fig. V-5), on $\phi_{\text{dry-out}}$.

References 44 through 48 pertain to organic-coolant technology and originated in a number of different countries. Reference 44 is particularly interesting in that it traces the

Fig. V-6 Typical sheath-temperature plot obtained from FLARE in dry-out test.⁴³

designation of the phenomena that result after the complete evaporation of the protective water film on the heat-transfer surface.⁴³ Figure V-5 illustrates the test section used in FLARE. Determination of the dry-out heat flux was made for a variety of conditions, and Fig. V-6 illustrates typical wall temperatures for one of the tests. This test was described as follows:

... At the start the test-element power (and heat flux) is being increased until indications of dry-out

evolution in thinking on the use of zirconium alloys in organic-cooled systems. In the late 1950's, preliminary experience had indicated that zirconium alloys were unacceptable for use in organic-cooled reactors due to hydriding. This led to increased study of sintered aluminum (SAP), and the work with zirconium was deemphasized. More recent information, however, has indicated that zirconium alloys may be usable in organic systems provided the

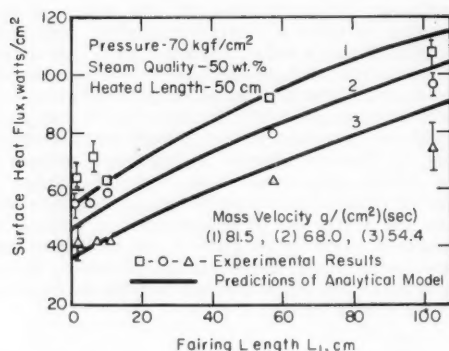


Fig. V-7 The effect of unheated length on the dry-out heat flux.⁴³

coolant impurities are carefully controlled. According to the reference:⁴⁴

The work to date in Canada has shown very clearly that the control of coolant purity is absolutely essential for an organic-cooled reactor. It has been shown further that coolant purity must never be allowed to get out of control since otherwise the subsequent clean-up will be long and difficult. Several years of operating in-reactor and out-reactor loops has proved that the control of coolant purity is no more difficult than in a water-cooled reactor. Provided the proper conditions are observed fouling will not be a problem, at least up to surface temperatures of 500°C, and zirconium alloys such as Zr-2.5% Nb show real promise for fuel sheathing and pressure tubes.

The important variables seem to be water, hydrogen, and chlorine. Water dissolved in the organic seems to have a beneficial effect in reducing hydriding in certain concentration ranges, as may be seen in Table V-8. Increased hydrogen concentrations increase the rate of hydriding at water concentrations less than 40 ppm, but, at "higher" water concentrations, the effect of hydrogen is said to be negligible.⁴⁴ Chlorine is thought to be the most important

promoter of zirconium hydriding, and Table V-9 illustrates the effect. Surface preparation has been shown to be important, and the effect of temperature appears minor in the temperature range 365 to 425°C.

Table V-9 EFFECT OF CHLORINE ON HYDRIDING OF ZIRCALOY-2 (REF. 44)

	No chlorine added	50 ppm C_2HCl_3 added initially
Temp., °C	375	370
Chloride in coolant at 370°C, ppm		
At start	0.8	8.2
At end (after 67 hr)	0.5	1.6
Hydrogen absorbed, $\mu g/(cm^2)(hr)$		
Pickled surface	0.39	9.4
Oxidized surface	0.01	2.1

The same importance of coolant purification was noted in the French studies.⁴⁶ These experimenters studied the effect of various impurities on fouling in terphenyl mixtures. A heated, stainless-steel test section was utilized, but portions of the loop were fabricated of mild steel. The study was directed toward the elimination, or control, of fouling and concluded that fouling could be avoided if the terphenyl used contained "few impurities and no oxygen."⁴⁶ Heat-transfer coefficients were determined during some of the experiments, and a correlation is given in the reference. At wall temperatures in excess of 500°C, a decrease in the heat-transfer coefficient was observed, and this was attributed to localized pyrolysis of the coolant at the surface.

References 47 and 48 discuss burnout heat fluxes for organic coolants. The former is on terphenyl mixtures, terphenyl-benzene, and terphenyl-diphenyl mixtures, whereas the latter is on a biphenyl-diphenyl methane eutectic, a biphenyl-diphenylozid mixture, and Dowtherm A.

Table V-8 EFFECT OF WATER ON HYDRIDING IN AUTOCLAVES⁴⁴
(400°C; Dissolved H_2 , 100 ml/kg)

Water content of organic, ppm	Hydriding rate, $\mu g/(cm^2)(hr)$					
	Zircaloy-2		Ni-free Zircaloy-2		Zr-2.5% Nb	
	As pickled	Preoxidized	As pickled	Preoxidized	As pickled	Preoxidized
40	0.5	0.032 ± 0.007	0.5	0.010 ± 0.005	1.0	0.12 ± 0.012
40 to 60	0.4	0.011 ± 0.002	0.280 ± 0.020	0.001 ± 0.001		0.001 ± 0.001
60 to 1000	0.045 ± 0.02	0.010 ± 0.01	0.025 ± 0.015	0.002 ± 0.002	0.025 ± 0.005	0.001 ± 0.001

Both references employ correlations of the form $\phi_{BO} = a + bG^c + d \Delta T G^e$ for their results, where ϕ_{BO} is the burnout heat flux, G is the mass flow rate, and ΔT is the subcooling. The remaining symbols are constants for a particular fluid.

Directions of Future Developments

The direction of development of the heavy-water-moderated power-reactor program in Canada is covered in Refs. 49 to 51. The near-future plans envision the construction of a 1000-Mw(e) station composed of two 500-Mw(e) plants of the CANDU type. This reactor would evolve from CANDU by means of the following improvements:⁵⁰

- Use of cold-worked zirconium-2½% niobium pressure tubes

- Increase of outlet temperature from 294 to 301.7°C

- Increase of fuel rating from a conductivity integral of 40 to 48 watts/cm

Other changes involve an increase in the pressure-tube diameter, a corresponding decrease in the number of pressure tubes, and, possibly, venting the containment pressure-relief system to a separate gas holder and heat absorber. Interestingly, the large reactor will be controlled by absorbers composed of liquid or gas arranged in vertical through tubes distributed throughout the reactor core, and by soluble poison dissolved in the moderator. The vertical through tubes will be subdivided into a number of separate chambers for flexibility. It thus appears that moderator-level control will not be used for the large reactors, probably for the reasons discussed in Sec. VIII of *Power Reactor Technology*, 7(3).

The longer term direction of the Canadian program is summarized as follows:⁵¹

For the water-cooled line itself, the following evolution appears reasonable:

Step 1—(Present Stage)—Pressurized D₂O with CANDU Type UO₂ Fuel Bundles Operating at a Rating of $\int k d\theta = 40$ W/cm.

For the same reactor, a cheaper fuel can be developed by using larger fuel elements operating at $\int k d\theta = 48$ W/cm.

Step 2—Boiling D₂O—Indirect Cycle—UO₂ Fuel. From 10% to 20% outlet quality, little change is expected in fuel geometry. From 15% to 30% quality, fuel and/or pressure

tube geometry may have to change to keep coolant distribution optimum.

Step 3—Boiling H₂O—Direct Cycle—UO₂ Fuel. When outlet qualities between 20% and 30% are shown to be practical, this reactor will be a strong contender.

Elsewhere, as in Canada, the key question in the future development of the heavy-water pressure-tube reactor is which coolant to use. Although the D₂O-cooled version of the reactor has been developed in Canada, its disadvantages with respect to steam-temperature limits, D₂O leakage, and D₂O inventory considerations are appreciated there, as is attested by the Canadian work on the H₂O-cooled and organic-cooled versions. Considering the work throughout the world, one concludes that the three alternate coolants, gas (CO₂), organic liquids, and H₂O in some form, will receive serious developmental tryouts. Of the coolant systems that have been considered seriously in the past, only sodium and direct-cycle boiling D₂O appear to have been eliminated.

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Section

VI

Power Reactor Technology

Heavy-Water-Moderated Pressure-Vessel Reactors

General Status

The heavy-water-moderated pressure-vessel reactors have had their most extensive development in Sweden, where the Agesta reactor was built, and in Norway, where the Halden Boiling Heavy Water Reactor (HBWR) is located. The Agesta Nuclear Power Plant (R3/Adam) was started in 1958, completed in 1962, and reached full power in March 1964. The HBWR has been operating since 1959. Further characteristics of both of these reactors are given in Table VI-1.

Heavy-water systems and fuel-handling equipment for the Agesta reactor are discussed in Refs. 1 and 2. Many of the details exemplify straightforward engineering practice, although some are peculiar to the reactor type. Among the latter is the careful preservation of cleanliness of the primary system:

Before the final assembly of the components at the production workshops, a very comprehensive and stringent cleaning ... was carried out, for the most part in specially arranged clean rooms where the personnel worked under special clean conditions. Cleaning was mostly effected by washing ... in alkaline water and by washing or wiping with acetone.

Pipes and components of carbon-steel were also used, to a limited extent in the heavy-water systems. After cleaning, these parts were rust-protected by a final rinsing in a bath of VPI dissolved in methanol.

After the final assembly, pipe connections and other access openings were carefully sealed with, for example, plastic covers. Large carbon-steel tanks were not rust-protected by means of VPI but

kept filled with a controlled over-pressure of dry nitrogen.

A number of basic designs for components in the Agesta heavy-water system are shown in Fig. VI-1. The centrifugal pumps were all canned pumps, whereas the D₂O piston type, feedback pumps were constructed with double seals around the piston shaft with provision for intermediate drainage. The latter principle, of double seals with D₂O recovery through leak-off pipes, is used generally for the nonhermetic closures.

A unique design feature of the Agesta fuel-handling equipment is the provision for transfer of fuel elements from the reactor D₂O to the storage pool, which contains H₂O. Under normal operation the transferred fuel element is gas cooled after its removal from the reactor and prior to its placement in the storage pool. Clean nitrogen is used as a coolant, and the evaporated D₂O is condensed by means of a Freon refrigeration system. In the event of failure of the gas cooling system, a backup spray of either D₂O or H₂O is provided. This spray system also provides cooling when the gas system is not being utilized. A simplified scheme of the cooling and drying circuit is shown in Fig. VI-2.

The second Swedish heavy-water reactor, the Marviken³ plant, is also of the pressure-vessel type but is cooled by boiling D₂O in natural circulation. The plant is now under construction near Stockholm and should be operational in 1968. Details of the design are given in Table VI-2. The flow path of the primary coolant,

Table VI-1 CHARACTERISTICS OF THE AGESTA AND THE HBWR

Reactor	Power	Coolant	Coolant pressure, psi	Fuel/cladding (first core)
Agesta	10 Mw(e); 55 Mw, space heating	Pressurized D ₂ O	500	UO ₂ /Zircaloy-2
HBWR	10 Mw, process steam	Boiling D ₂ O	400	Uranium/aluminum

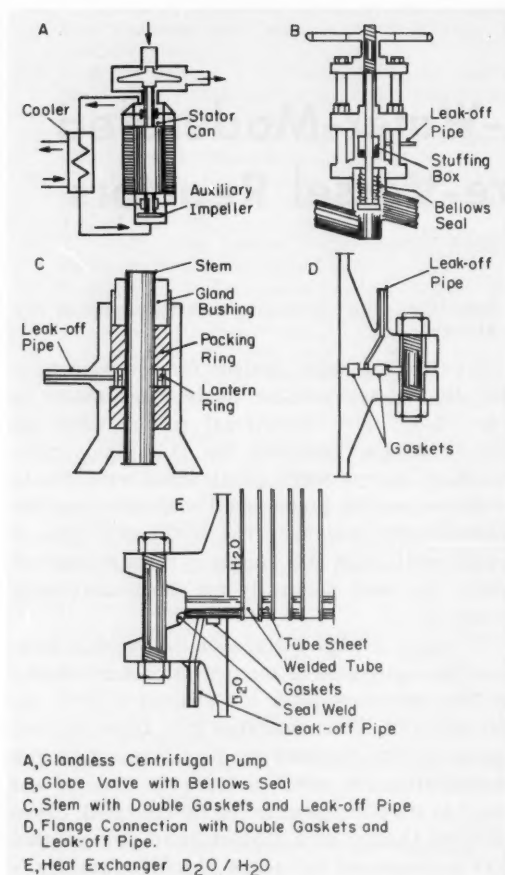


Fig. VI-1 Basic designs for components in the Agesta heavy-water system.¹

Table VI-2 DESIGN DATA FOR THE MARVIKEN BHW⁴

Reactor type	BHWR
General	Direct
Cycle	Natural
Circulation	Internal,
Superheating	when provided
Fuel	UO ₂
Fuel changing	Onload, continuous
Data	
Net electrical output, Mw(e)	200
Net thermal efficiency, %	34.0
Mean moderator temperature, °C	163
Core volume	
Unreflected, m ³	64.1
Reflected, m ³	94.6
Inner diameter of pressure vessel, m	5.22
Design pressure, bar	57.5
Working pressure, bar	49.5
Pressure-vessel steel	Low alloy (A-302B)
Wall thickness (excluding S.S. lining), mm	76
Outlet steam temp., °C	470
Average enrichment, %	~1.35
Average burnup, Mwd/ton of U	13,000
Fuel for boiling channels	
No. of channels	147
No. of rods per cluster	31
Rod diameter (UO ₂), mm	12.5
Can material	Zircaloy-2
Can thickness, mm	0.55
Superheater elements* (2nd charge)	
No. of channels	32
No. of rods per cluster	48
Rod diameter (UO ₂), mm	~11.5
Currently assumed can material	20 Cr-35 Ni
Currently assumed wall thickness, mm	0.4-0.25†
Approx. fin height, mm	0.1
Max. surface temp. (hot spot), °C	524

*Current assumption.

†Future changes.

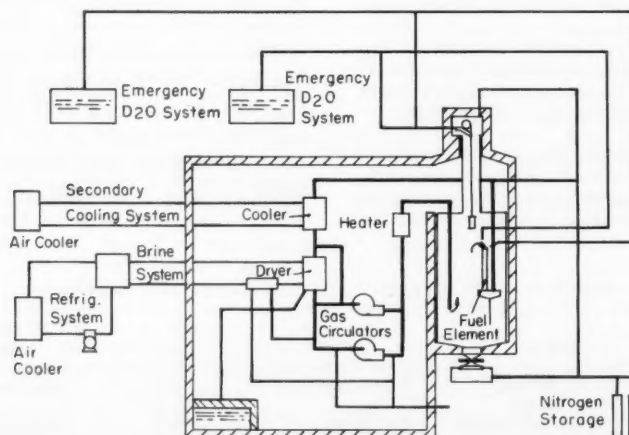


Fig. VI-2 Simplified scheme of the Agesta cooling and drying circuit.²

which is quite unusual, is shown in Fig. VI-3. Water from the feedwater heaters is pumped into the moderator space at a temperature of about 250°F. The core is constructed of 179 Zircaloy-2 shroud tubes hydraulically isolating the moderator region of the core from the fuel elements. The moderator water therefore flows to the top of the moderator space and exits at a temperature of about 430°F. At the top of the core, the moderator mixes with the liquid portion of the two-phase mixture that cools the fuel elements, and the resulting coolant flows to the downcomer and thence to the fuel elements. The initial operation of Marviken will be with saturated steam.³ Thirty-two of the shroud tubes will be left empty to serve as flow channels for the saturated steam; the remainder will contain fuel elements. The turbine contains an inlet blading system suitable for saturated-steam operation and interstage moisture separation, but the inlet blading can be modified for operation with superheated steam. When operating on the saturated-steam cycle, Marviken generates 140 Mw(e). The rated power of 200 Mw(e) is attained only through incorporating superheating fuel elements in the 32 empty shroud tubes.

Operation with the relatively cool moderator is done to improve burnup for a given enrichment.³ High-pressure tests in a mockup of the reactor was 0.5% for steam carry-under and negligible carry-over of water. Figure VI-4 shows an overall arrangement of the reactor plant. The containment is based on the pressure-suppression principle.

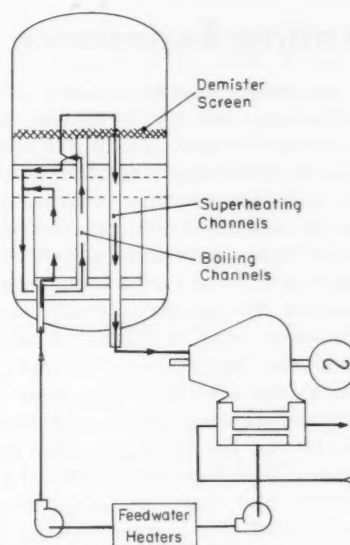


Fig. VI-3 Flow schematic for the Marviken reactor.⁴

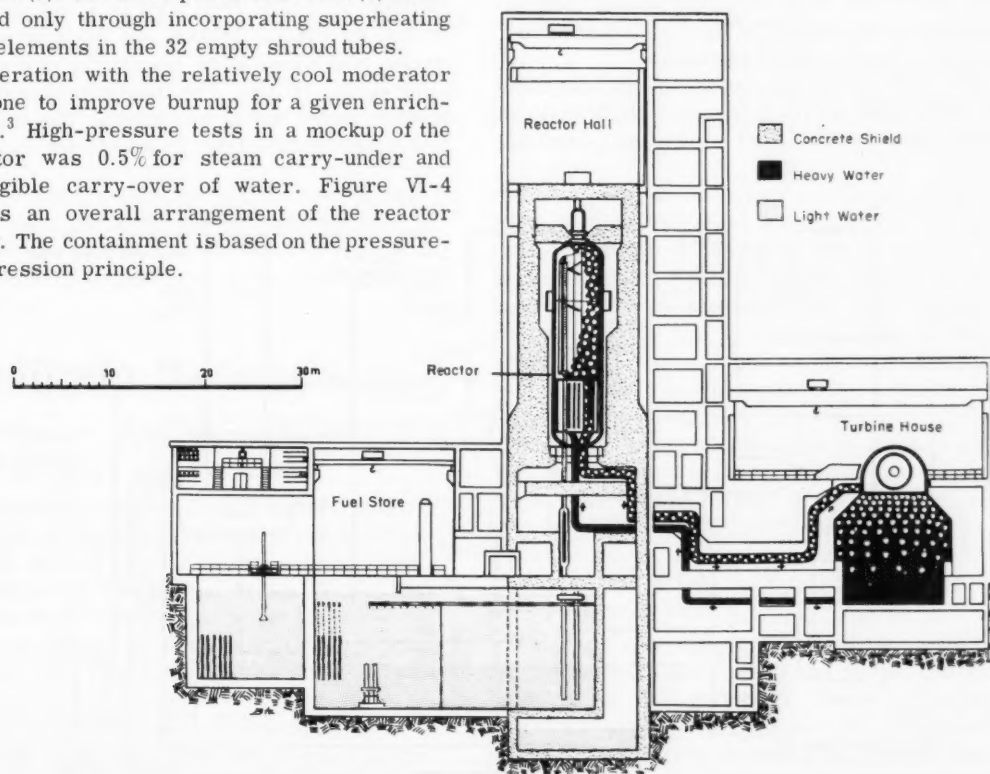


Fig. VI-4 Schematic of the Marviken reactor.

tilation air flow rate and absolute humidity. Inventory measurements were also made, and they corresponded "reasonably well" with the tracer results. The loss of heavy water and the reactor pressure history for the HBWR are shown in Fig. VI-6. The heavy-water inventory for this reactor is 14.5 tons.⁷ The loss of 2500 kg of D_2O over a 20-month period (Fig. VI-6) represents a leakage rate of about 10% per year.

helical wire wrap is not applicable to this type of design because of the 3- to 5-mm spacing between rods. The Agesta design provides for spacing and intermediate support points by subdividing the 3-m fuel rod into four subelements and using spring-loaded flexible plate spacers located at the fuel-element junctions.

The Marviken boiling fuel elements are composed of rods built up of two half-length sub-

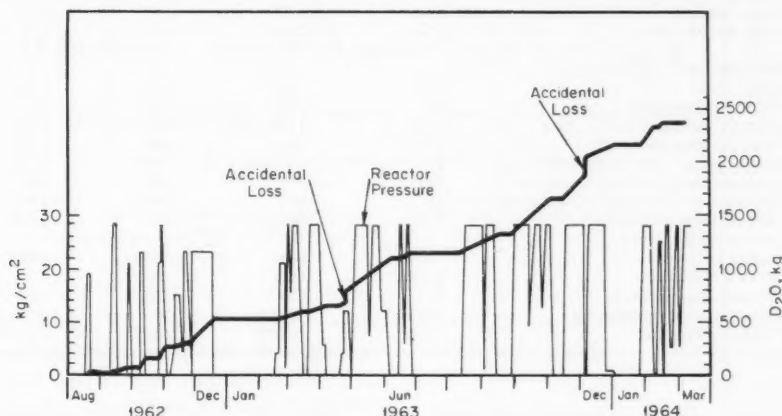


Fig. VI-6 Loss of heavy water in the HBWR and reactor pressure history.⁷

At the start of operation with the second core, the isotopic purity of the D_2O was 99.75%. By the end of 1963 it had fallen to 99.68%. This represents an inleakage of about 10 kg of H_2O to the system.

Significant Developments

Fuel Elements, Fuel Materials, and Cladding

The Swedish experience with fuel and cladding materials is contained in Refs. 8 and 9. Reference 10 is a Norwegian contribution on aluminum alloys as cladding materials. Table VI-4 gives fuel data for the Agesta and Marviken reactors and the pressurized heavy-water reactor (PHWR). The PHWR is a design study and will be discussed shortly.

The Agesta fuel bundle is relatively open, in the sense of having a large rod-to-rod spacing. This is to ensure adequate coolant velocity in natural circulation and also to provide a slightly negative void coefficient. It is said⁸ that the

rods. A midbundle spacer positions the sub-rods, and the reference mentions that the tendency for steam bubbles to anchor on the spacer grids has been a problem with respect to localized corrosion and overheating. The rod-to-rod spacing in the Marviken is about 9 mm. This is maintained by the spring type spacers. For rigidity the spacers are anchored to a central, supporting Zircaloy tube that apparently is unfueled. In-pile test elements of Swedish design were run in the Canadian NRX reactor and in the VBWR. The highest oxide temperatures were determined by postirradiation examinations to be in the pellet immediately downstream of the spacer, and this is attributed to steam-bubble anchoring.⁸

The corrosion and hydriding results of Swedish experiments on Zircaloy are typical of those observed in other countries. The shroud in the Marviken reactor is a fixed piece of core hardware; thus the requirement of a long in-core lifetime does exist, although the consequences of a failure are not as serious as the failure of a pressure tube in a pressure-tube reactor.

Table VI-4 FUEL DATA FOR SWEDISH POWER-REACTOR FUEL ELEMENTS⁸

	Agesta, 55 Mw(t) + 10 Mw(e)	PHWR, 360 Mw(e)	Marviken	
			Boiling element, 140 Mw(e)	Super- heating element, 200 Mw(e)
Total heat release in fuel, Mw	60	1134	471	122
Reactor pressure, bars	33.3	74	49.5	49.5
Fuel form	UO ₂ (sint.)	UO ₂ (sint.)	UO ₂ (sint.)	UO ₂ (sint.)
Enrichment, %	Natural	1.5	1.3	1.6
Active fuel-element height, mm	3047	4455	4420	4220
Free space in rod, mm	10	65	100	150
Fuel rods per assembly	4 × 19	2 × 49	2 × 30	2 × 48
No. of assemblies	140	177	147	32
Total UO ₂ weight, tons	18.5	33.7	25.6	7.0
Can	Zircaloy-2	Zircaloy-2	Zircaloy-2	S.S.
Canning-tube inside diameter, nominal, mm	17.2	10.5	12.6	11.6
Can wall thickness, mm	0.7	0.51	0.6	0.4
UO ₂ pellet diameter, nominal, mm	17.0	10.4	12.5	11.5
Mean heat loading, watts/cm	78.5	300	227	180.6
Hot-channel factor (including transient)	4	2	2.99	2.72
Max. transient heat loading, watts/cm	314	600	680	491
Max. heat flux on can surface, watts/cm	54	164	156	126
Mean temperature of the fuel, °C	500	900	800	1200
Max. center temperature (including transient), °C	1325	2450	2760	2670
Coolant/UO ₂ area ratio	1.13	16.4	3.52	1.92
Coolant inlet temperature, °C	205	254	255	263
Coolant outlet temperature, °C	220	274	263	470
Mean burnup, Mwd/ton	3600	16,000	13,000	13,000

The use of aluminum as a cladding is being studied primarily as an in-pile experiment.¹⁰ Prototype elements were scheduled to be irradiated in the HBWR in late 1964.

Water Chemistry

Reference 11 is a report on the water-chemistry research at the HBWR. The report covers reactor experiments that were designed to study conditions under which corrosion can be reduced to the point where operation is possible without full-flow condensate purification. This involves a search for conditions under which steam will be produced with a low oxygen content and the alkalinity of the recirculating water will be preserved. Reducing the in-core dissociation of D₂O should also reduce the recombiner load. However, this particular point was not mentioned in the reference.

Figure VI-7 shows the oxygen concentration in the HBWR steam as a function of power, with no additions to the coolant. The D₂ was found in stoichiometric amounts. When D₂ was added to the subcooled feedwater, the net decomposition of D₂O was considerably lessened, as is shown in Fig. VI-8. This figure shows the effect of D₂ addition at various pressures, various locations, and several different powers. Ammonia

(as ND₃) was added to the coolant, and it was shown that the radiolysis of ammonia into nitric acid could be prevented by maintaining the D₂ concentrations at or above 0.3 to 0.4 ml per kilogram of heavy water. The radiolysis of ammonia into nitrogen and deuterium did increase the ¹⁶N content of the steam, however.

Control and Instrumentation

The Agesta reactor is equipped with top-mounted, hydraulically actuated control-rod drives.¹² A stepping action is accomplished by

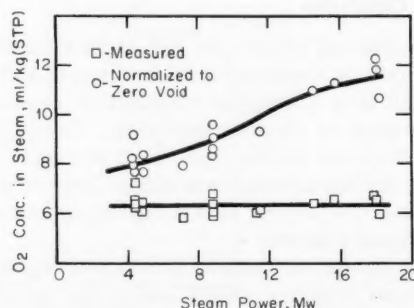


Fig. VI-7 Oxygen concentration in steam vs. steam power at 28 ata.¹¹

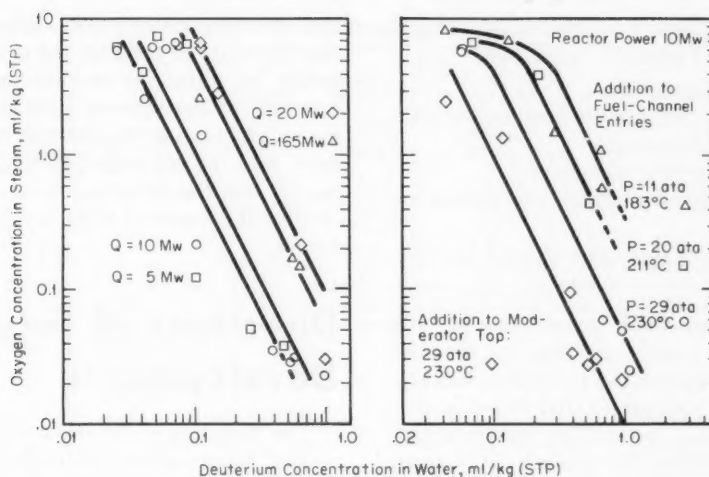


Fig. VI-8 Oxygen concentration in steam vs. deuterium concentration in water entering the fuel channels.¹¹

means of two chucks. One chuck normally holds the rod in a given position, and the other grasps the control rod and moves a small distance up or down (10 mm) in response to hydraulic pressure. The Agesta control rods are of silver-indium-cadmium alloy clad with type 304L stainless steel and are tubular in shape.

The Marviken reactor will be equipped with rather unusual control-rod drives. The reactor (Fig. VI-4) is designed for on-power refueling. The entire control-rod drive is to be located within the pressure vessel, immediately above the core, and will be hydraulically actuated. The drive will be positioned by means of the fuel-handling machine and held in place only by the weight of the drive itself. The water, which serves as the hydraulic fluid, is fed through pipes penetrating the pressure-vessel wall and is introduced into the drive by means of piston-ring type seals. The drives are located in the steam atmosphere above the core, but the drive mechanism will be filled with water to control the drop speed upon scram. The position-indicating system is based on ultrasonic location of a reflector plate at the lower end of a control rod. A transducer, at the bottom of the control-rod channel, emits ultrasonic pulses, and the time interval between their transmission, reflection, and absorption back at the transducer gives the location of the control rod.

The in-core instrumentation for the HBWR is detailed in Ref. 13. Most of the information has

been presented in *Nucleonics*¹⁴ and will not be repeated here.

Pressure Vessels

The design and fabrication of the Agesta and Marviken pressure vessels are discussed in Ref. 15. The design details of these vessels, and that of a larger reactor to be discussed shortly (BASHFUL-600), are given in Table VI-5. The Agesta vessel has a cylindrical shape with an ellipsoidal bottom dome and a flat lid. The cylindrical portion is roll clad with stainless steel and has an upper flange fabricated of forged parts welded together and clad with weld-deposited stainless steel. The Marviken vessel is quite different from that of the Agesta and is shown in outline form in Fig. VI-4. The

Table VI-5 SURVEY OF REACTOR VESSEL DATA¹⁵

	Agesta PHWR	Marviken BHWR	BASHFUL-600
Design pressure, bars	40	57.5	80
Design temperature, °C	251	272	293
Vessel inside diameter, m	4.5	5.22	6.6
Cylindrical wall thickness, m	65	76	120
Bottom dome-wall thickness, mm	65	70	175
Vessel steel type	C-Mn	C-Mn-Mo	9% Ni steel
Vessel height, m	9.5	23.96	26.75

bottom dome, which is hemispherical in shape, is penetrated by 87 nozzles, as follows:

- 32 superheater nozzles
- 34 nozzles that indicate the positions of control rods by ultrasonics
- 12 nozzles for neutron detectors
- 8 nozzles for startup heating and coolant inlet
- 1 nozzle that is a transport channel for control rods and fuel elements

The entire internal shell surface is clad with stainless steel by overlay welding, and the shell is penetrated by nozzles for feedwater, control-rod-drive piping, pressure-relief valves, emergency cooling piping, and canning leak-detection piping. The hemispherical upper head is penetrated by three nozzles for television cameras and the large central nozzle supporting the charging machine. The vessel closure system is unique and is illustrated schematically in Fig. VI-9. The design is such that all forces acting in the fixed flange rings pass through one point, thus giving no moment in the ring.

The pressure vessel for the 600-Mw(e) BASHFUL reactor, which is an advanced design of a boiling heavy-water-moderated reactor, has an inside diameter of 6680 mm and a total height of 24,750 mm. The vessel would be shipped to the

site in at least two pieces. After installation of the internals at the site, the parts of the vessel would be joined by welding and locally heat-treated. At the present time no welding electrode has been developed that is fully satisfactory with the 9% nickel steel proposed for the vessel, but the reference states that such electrodes are expected to be developed in the near future.

Directions of Future Developments

The direction of development of the pressure-vessel heavy-water-moderated power reactor appears to be largely dependent on the future program in Sweden, as well as on the Norwegian work, although there apparently is also some activity on the reactor type in Germany.¹⁷ Sweden, probably because it is relatively well endowed with water power, anticipates a slow buildup of nuclear capacity in the near future but a rapidly accelerating one later. Reference 18 assumes that Sweden will have an installed nuclear capacity of 200 Mw in 1970, 1000 Mw in 1975, and 4000 Mw in 1980. The figures seem to indicate that, when nuclear energy is finally

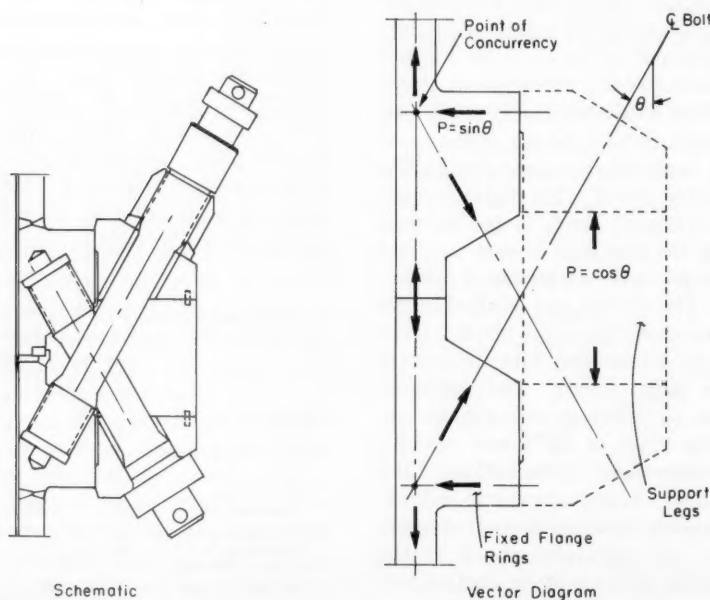


Fig. VI-9 Details of Marviken vessel-closure flange.¹⁵

Table VI-6 PARAMETERS CHARACTERIZING ADVANCED SWEDISH REACTORS¹⁸

	Marviken	BHWR*		PHWR	BASHFUL†
		Natural U	Enriched U		
Thermal efficiency	0.287	0.314	0.314	0.283	0.371
Specific power, kw/kg of U	18.0	18.0	27.0	35.6	24.0/41.2
Enrichment, wt.% ²³⁵ U					
Initial	1.00	0.71	1.17	1.50	1.05/2.17
Final	0.25	0.25	0.25	0.40	0.2/0.92
Rod diameter, mm	12.5	14.0	11.5	10.4	11.5/8.5
Plutonium, g/kg of U	5.6	4.7	6.6	6.0	6.3/6.9

*The BHWR is a 300- to 500-Mw(e) reactor producing saturated steam.

†BASHFUL is a reactor similar to Marviken in the 400- to 1000-Mw(e) power range. The numerator of the fractional entries pertains to the parameters of the boiling region, and the denominator pertains to the superheating region.

needed, it will be badly needed. This is substantiated in the reference, which states that domestic sources of fossil fuels in Sweden are lacking.

The Marviken reactor represents the 1970 commitment approximately. Beyond this date the reactor type to be built is open; pressurized heavy-water reactors (PHWR),¹⁹ saturated boiling heavy-water reactors (BHWR), and boiling and superheating reactors of the Marviken type (BASHFUL)¹⁸ are all under study. Reference 18 is a discussion of the economics of these various types, and some of the proposed design parameters are given in Table VI-6. Fuel costs are given in Table VI-7. Table VI-7 illustrates the importance of on-power refueling, since this is said to be responsible for the difference in cost between the PHWR and the remaining reactors. The PHWR costs were arrived at by assuming multizone, batch loading, whereas the remaining plants were assumed to be continuously refueled. The penalty can be seen in the enrichment values given for the PHWR, as con-

trasted to the remaining plants. The processing of fuel appears to be of marginal value, as Table VI-7 shows that the cost of processing about equals the plutonium value.

The Swedish fuel-cycle philosophy is now characterized as follows:¹⁸

Use of slightly enriched uranium for normal operation but possibility of optional operation with natural uranium.

Once-through uranium cycle without reprocessing of spent fuel but with possibility of future recycling of plutonium in a self sustained cycle.

The option for use of natural uranium is somewhat peculiar to Sweden but is of general interest. If 4000 Mw(e) is in fact the installed capacity in 1980 and if the annual increase is 800 Mw(e), the net import requirements for natural uranium would amount to \$8 million per year, compared to \$13 million for enriched uranium. If the nuclear electricity were being generated with conventional BWR's, the corresponding figures, as given in Ref. 18, would be \$40 million per year for Zircaloy-clad fuel and \$63 million

Table VI-7 FUEL COSTS* FOR DIFFERENT REACTOR TYPES¹⁸

Cost item, mills/kw-hr	Marviken	BHWR		PHWR	BASHFUL
		Natural U	Enriched U		
Fuel consumption	0.51	0.39	0.63	1.04	0.67
Manufacturing	0.88	1.23	0.89	1.10	0.79
Fixed	0.20	0.13	0.13	0.14	0.13
Processing	0.29	0.44	0.26	0.30	0.23
Credit on uranium	0	0	0	-0.04	-0.09
Credit on plutonium	-0.43	-0.55	-0.46	-0.47	-0.38
Once-through (items 1 to 3)	1.59	1.75	1.65	2.28	1.59
Net credit (items 4 to 6)	-0.14	-0.11	-0.20	-0.21	-0.24
Total cost	1.43	1.64	1.45	2.07	1.35

*Basic assumptions: burnups, 15 Mwd per kilogram of uranium, except for natural uranium, which is 9 Mwd per kilogram of uranium; feed cost, \$8 per pound; separative work, \$30 per kilogram of uranium; interest rate, 7%; fixed costs at 6000 h/g and 60% of initial core; processing, \$30 per kilogram of uranium; plutonium credit, \$8/g total.

for stainless-clad fuel. These numbers might be important in considerations involving trade balance or emergency situations, but the reference assigns them a "minor importance." Nevertheless, the requirement that the reactor be capable of operation with natural uranium is allowed to compromise the optimization of the reactor fueled with enriched uranium, and this is noted in the reference.¹⁸

Czechoslovak Nuclear Power Station

The remaining heavy-water power reactor of the pressure-vessel type is the gas-cooled reactor of the Czechoslovakian nuclear power station. Few design details were presented at the conference, but coverage had been reasonably thorough at the Second Geneva Conference. Reference 20 reports on some design and construction problems of the reactor and discusses the charge-discharge machine, control-rod tests, top biological shield design, steam-generator tests, main blower tests, and steam-turbine design. The plant has been scheduled²¹ to be critical in 1965. No other dates are given to suggest that this time schedule will not be met, although it is mentioned in Ref. 20 that the construction time has been "elongated."

The pressure vessel for the Czechoslovakian nuclear power station is discussed in Ref. 16 and is shown in Fig. VI-10. The figure does not

illustrate the upper head details, but there are approximately 200 nozzles through the head for control and refueling. The reactor is D_2O moderated, and the CO_2 coolant is piped through 12 inlet and 12 exit nozzles. The reference contains much information on stress analysis, material properties, and test results that would be of interest to the specialist. The pressure-vessel steel is designated "CSN 13030," and is composed of <0.20% carbon, 1.10 to 1.14% manganese, <0.10% cobalt with titanium and aluminum added "... for increasing of structural stability and finegrainedness ..."¹⁶ The reference describes the manufacture of a shortened version of the pressure vessel. The vessel is the same as the one shown in Fig. VI-10 but does not include some of the ring sections.

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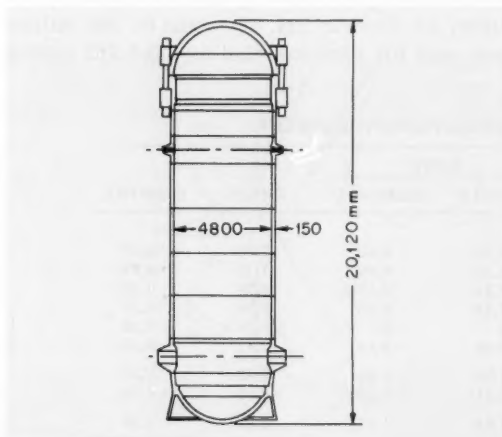


Fig. VI-10 Pressure vessel for the first Czechoslovakian power station.¹⁶

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Section VII

Power Reactor Technology

Gas-Cooled Clad-Fuel Reactors

By Walter Mitchell III

General Status

The Geneva information on gas-cooled reactors that employ metallic fuel jackets deals almost exclusively with reactors developed in the United Kingdom and France. Consequently, this article reviews the status, the significant developments, and the indicated directions of future development in terms of the British and French programs.

Some basic information on specific reactors of the type is presented in Table VII-1. The reactors, all of which are CO₂ cooled and graphite moderated, are classified as "British designed" or "French designed," but the reactors in the former category are not all located in the

British Isles. Although the majority of the stations listed are multiple-reactor installations, as indicated in the first column of Table VII-1, the net electric outputs listed pertain to single reactors. The date of power operation for a particular station gives the year in which the first reactor in a multiple-reactor installation began or will begin operation.

The large number of British reactors of the CO₂-cooled graphite-moderated type built or being built and the steady progression in reactor output are striking results of the British program, which has up to now concentrated heavily on this reactor type. The improvements in the type, as indicated by the parameters listed in Table VII-1, illustrate rather vividly the technological advances that occur as a re-

Table VII-1 GENERAL INFORMATION ON GAS-COOLED CLAD-FUEL REACTORS¹⁻¹⁸

Station (number of reactors in parentheses)	Date of first reactor on power	Net electric output per reactor, Mw	Net efficiency, %	Primary coolant nominal pressure, psia	Primary coolant temperature at reactor outlet, °F	Fuel rating, Mw(e)/ metric ton of U	Capital cost, \$/kw
British-designed reactors							
Calder Hall (4)	1956	54	23.5	100	652		
Chapelcross (4)	1959	54	23.5	100	652		
Berkeley (2)	1962	138	24.4	140	653	0.59	521
Bradwell (2)	1962	150	28.2	147	739	0.63	492
Latina (1)	1963	200+	28.4	197	739		
Hinkley Point (2)	1964	250	26.4	195	705	0.72	420
Hunterston (2)	1964	150	28.3	161	744	0.60	
Trawsfynydd (2)	1964	250	29.4	246	735	0.85	384
Dungeness (2)	1965	275	32.9	284	770	0.92	319
Tokai (1)	1965	149	26.8	223	739		
Sizewell (2)	1965	290	30.5	282	770	0.91	300
Oldbury (2)	1966	300	33.6	365	770	1.02	302
Wylfa (2)	1968	590	31.5	400	770	1.00	280
AGR (1)	1962	27.3	27.3	285	977		
French-designed reactors							
G2 (1)	1959	40	16.0	215	644		
G3 (1)	1960	40	16.0	215	644		
EDF-1 (1)	1963	70	23.3	406	670		
EDF-2 (1)	1964	200	23.6	385	716	0.80	
EDF-3 (1)	1966	480	30.8	435	770	1.14	240
EDF-4 (1)	1968	480	29.1	435	752	1.08	

sult of the research engendered by a large building program and the experience gained through reactor operations. The French program, although considerably smaller than the British effort, has resulted in rather remarkable advances also. Indeed, the predicted performances of the French reactors presently under construction are fully as impressive as those of the British. For example, a comparison of the data in Table VII-1 for the Sizewell and EDF-3 plants, which are scheduled for on-power operation in 1965 and 1966, respectively, shows that the bulk outlet gas temperatures from the reactors are the same. Pressure and fuel rating in EDF-3 exceed the comparable values in Sizewell by a large margin. Comparisons of cost estimates from different countries can be misleading, but, if the estimates for Sizewell and EDF-3 are comparable, they indicate a considerable advantage for the French design, perhaps because of the larger reactor output.

That the British are proceeding with such a vigorous construction program is not surprising in view of the fact that, even under conservative assumptions, nuclear power in that country is expected to command a place in the generating system on solely economic grounds in the period 1970 to 1975. An indication of the relative costs of nuclear and conventional power in Great Britain is given in Ref. 9 by the statement that the replacement generation cost per day for Bradwell was \$75,500 in January 1963 and \$36,400 in July 1963. (The replacement cost is the incremental fuel cost per day of the next available stations in the order of merit needed to replace Bradwell's 300 Mw.) Although one would expect the January figure to be high, since this is a peak-load time of the year and the conventional plants available for service are the old and expensive plants, the July figure represents a period of low load during which time it appears reasonable that some fairly good conventional plants would be available for use. The January 1963 replacement generation cost is equivalent to a fuel cost of over 10 mills/kw-hr. It is interesting to note that even the very large conventional power stations that will be commissioned in Great Britain during the period 1965 to 1970 are estimated to have fuel costs¹³ of between 4.1 and 5.8 mills/kw-hr.

An interesting trend has developed in the use of the prestressed-concrete pressure vessel for

this reactor type. In the British program, all reactors down through Sizewell use metal pressure vessels, whereas Oldbury and Wylfa use prestressed concrete in conjunction with the "integral-plant" arrangement described later in this article. The French, despite (or perhaps because of) previous experience with concrete vessels in the G2 and G3 reactors, reverted to metal enclosures for the EDF-1 and EDF-2 reactors. Then, for EDF-3, they changed again—to a concrete version of the conventional pressure vessel, i.e., a vessel that encloses only the reactor proper. Finally, in the EDF-4 plant, the concrete-vessel integral-plant concept is used. Thus it appears that the evolution of the reactor type in the United Kingdom and France, after having proceeded along somewhat dissimilar paths, has now reached a point in time at which the gross features of the plants under construction are comparable. In the course of this evolution, it seems that the British may have displaced the French as the boldest exponents of the concrete pressure vessel, since they expect to put the integral concept into service in 1966 in the Oldbury plant, two years before EDF-4.

The large gas-cooled reactors have proved their reliability through the years. For some time now, utilization factors of 90% and greater have been common for long periods in the operating history of the Calder Hall reactors, and, as shown in Table VII-2, even the relatively young Berkeley and Bradwell plants are exhibiting a satisfactory degree of reliability. The "utilization factor" is defined as

$$\frac{\text{Units sent out} \times 100}{\text{Design sent-out capacity} \times \text{time}} \%$$

The performance of Berkeley and Bradwell during the period from startup to February 1964 is presented in Table VII-3 in terms of the utilization factor.

The reactors listed in Table VII-1, with the exception of the Advanced Gas Cooled Reactor (AGR), are fueled with natural-uranium metal clad in a relatively low-temperature alloy. The fuel elements are massive, to the extent that the entire cross section of a process channel contains only a single fuel element, and the specific power and power density are consequently low. These characteristics result from the use of metallic uranium, since the massive elements not only promote the attainment of the

Table VII-2 PERFORMANCE OF BERKELEY AND BRADWELL STATIONS DURING PEAK LOADING OF U. K. CENTRAL ELECTRICITY GENERATING BOARD SYSTEM^a

	Berkeley		Bradwell	
	Nov. 1962 to Feb. 1963	Nov. 1963 to Feb. 1964	Nov. 1962 to Feb. 1963	Nov. 1963 to Feb. 1964
Design sent-out capacity, Mw	275	275	300	300
Maximum sent-out capacity achieved, Mw	231	241	278	311
Units sent out, kw-hr $\times 10^{-6}$	528	557	552	807
Utilization factor, %	67	70	64	93
On-power dates	Reactor 1, June 1962 Reactor 2, Oct. 1962		Reactor 1, June 1962 Reactor 2, Nov. 1962	

Table VII-3 ANALYSIS OF PERFORMANCE FROM STARTUP TO FEBRUARY 1964, BERKELEY AND BRADWELL STATIONS^a

	Berkeley, %		Bradwell, %	
Utilization factor	52		60	
Utilization losses				
Output restrictions				
Nuclear fuel temperature	4.8		5.8	
Nonnuclear pressure drop in superheaters	8.0		0	
Plant nonavailability				
Nuclear				
Reactor trips	0.4		0.4	
Spurious reactor trips	1.1		0.6	
Flux scan tube failure	0		1.8	
Absorber, fuel, etc., movements off load	2.6		2.1	
Fuel failures	0.7	4.8	0.4	5.3
Nonnuclear				
Modification of superheaters	9.6			
Turbines, boilers, etc.	6.3	15.9	7.0	7.0
Planned maintenance	3.6		9.0	
External causes	0.2		0.3	
All other causes (load changing, startup, fuel integrity tests, etc.)	10.7		12.6	
	100		100	

necessary reactivity but also keep the fabrication component of the fuel cost at a reasonable level, despite the short exposure lifetime. The logical development of the reactor type, as embodied in the AGR, is a reactor capable of developing considerably higher gas temperatures and much higher power density, along with much extended fuel life. This is accomplished principally through the use of different fuel and cladding materials but requires the use of enriched fuel. Details of the AGR, in which the fuel consists of slightly enriched-uranium dioxide pellets encased in thin stainless-steel cans, assembled in clusters, are in Ref. 17. Fuel loading of the AGR commenced^b in August 1962, and since February 1963 the reactor has

operated at or above its designed power. The designed gross electrical generation of 33 Mw was achieved at startup, and within weeks the output was increased to 37 Mw, the limit of the installed generating plant. The experimental nature of the AGR has required the reactor to be shut down on several occasions, but, despite these periods of shutdown, the plant load factor (utilization factor) for the first year of operation has been a very satisfactory 74.1%. The plant availability factor, defined as the proportion of time for which the reactor was operating at full power, or for which the plant was fully available but was shut down or operated at reduced power for experimental purposes, has been 86.6%. These values, as well as the reasons for losses of equivalent full-power time, are shown in Table VII-4.

In summarizing the current status of the gas-cooled graphite-moderated reactor, one can really state only the obvious things: that a large number of the reactors have been built; that the reactors which have been built have operated satisfactorily and have proved that

Table VII-4 CAUSES OF REACTOR OUTAGE OF THE AGR^b (FEBRUARY 1963 TO FEBRUARY 1964)
(Losses As Percent of Equivalent Time at Full Power)

Planned annual maintenance and inspection, %	4.5
Unplanned maintenance, %	
Control-rod sleeve faults	4.6
Electrical/mechanical faults	2.8
Instrument faults	0.03
Human errors	1.5
Total operational losses, %	13.4
Operational availability, %	86.6
Off load or reduced power handling of irradiation experiments, %	11.7
Power reduction for physics experiments, %	0.8
Total experimental losses, %	12.5
Plant load factor, %	74.1

the type is reliable; that, typically, original performance capacities are exceeded by technological improvements which are incorporated in the existing plants; that the advanced reactors represented by the British AGR appear to have a sound basis for future development; and, finally, that the development of the technology has proceeded as might have been expected. Thus the very considerable gains that can be made by choosing a reactor type and building and developing it intensively have been demonstrated. If the type still appears to be less than economically competitive in countries like the United States, it is not because of any failure to equal or exceed the early expectations but because of inherent characteristics of the type.

Fuel Elements

The Magnox reactor, which is defined as a graphite-moderated gas-cooled reactor fueled with metallic uranium in Magnox cans, is the "standard" reactor discussed here. The cost reductions that have been achieved in the development of this reactor type are discussed in a paper by Smith et al.⁷ which, although written to apply to those reactors developed in the United Kingdom, serves as an excellent guide to the directions of development and remaining areas of exploitability of the general reactor type. The comments that follow are taken largely from Ref. 7.

The primary factors in the cost reductions which have been achieved are increases in unit output, improvements in layout and detailed design, and increases in fuel rating. Since the fuel electrical rating (electrical megawatts per metric ton of uranium) determines the total weight of fuel in a plant of given capacity, it is a significant parameter in the cost of the first fuel charge and also influences strongly the reactor dimensions, which, in turn, are reflected in the cost of the reactor and its associated equipment. The fuel electrical ratings can be increased by increasing fuel heat ratings, by increasing reactor coolant temperatures and steam-cycle efficiencies, or by lowering blower power. The achievement of a net gain by any of the above three means while keeping materials constant requires improvement of the fuel-element heat-transfer performance and/or the coolant heat-transport performance; in the gas-cooled reactor, the

most straightforward approach to such a gain is an increase in the coolant pressure. The trend toward higher pressures is illustrated in Fig. VII-1 for a majority of the Magnox reactors designed and built to date. The prestressed-concrete pressure vessels developed for the very large Magnox reactors have enabled the designer to select coolant pressure on the basis of economic optimization. All the high-output plants shown in Fig. VII-1 (Oldbury, Wylfa, EDF-3, and EDF-4) utilize the concrete vessel, and the attendant increase in fuel electrical rating is evident.

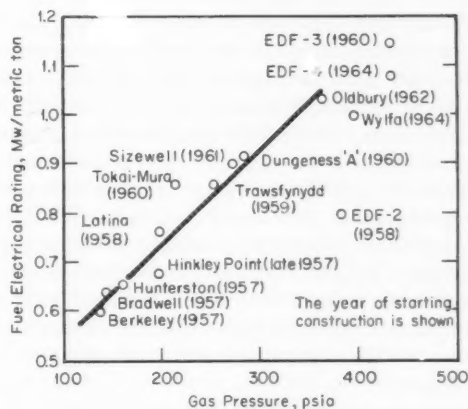


Fig. VII-1 Fuel ratings and gas pressures of various reactor designs.^{3,7}

As stated in Ref. 7: "The scope for increasing thermal efficiency of Magnox stations is probably fully exploited, so attention is now focused on increasing the fuel heat rate." Much of the effort on fuel and fuel-element development has been directed toward finding designs that can yield the higher heat ratings simultaneously with a fuel-element life expectancy that is at least as long as the reactivity lifetime afforded by natural uranium.

Fuel Materials

Certainly one of the most significant developments in the Magnox-reactor fuel program has been the production of "adjusted" uranium by the British. In the earlier days of the Magnox reactor, core life was considered to be determined by the swelling in the natural-uranium metal fuel. It is now considered that the behav-

ior of the uranium is not a limitation in the operation of the Magnox fuel elements and that the lifetime limit will be determined by the available reactivity. Basically, adjusted uranium is uranium containing 400 to 1200 ppm aluminum and about 300 ppm iron and 600 ppm carbon. These additions of material produce the greatest individual effect in reducing swelling, but, in order to obtain the effect, it is necessary to quench the uranium from the beta phase and subsequently anneal it in the alpha phase. In uranium produced by this method, the fractional volume increases are remarkably small and are unaffected by fission rate (for a given total burnup). British data¹⁸ indicate that the adjusted uranium is superior to uranium-molybdenum alloy in its swelling behavior and does not have the high-neutron-absorption penalty of the molybdenum alloy. Additionally, the molybdenum content of the uranium-molybdenum alloy presents a considerable problem in the chemical processing stage of the fuel cycle. Figure VII-2 illustrates the relation between the volume increases in uranium and uranium alloys. The adjusted uranium has been tested in thermal-cycling conditions of the amplitude and frequency expected in the Magnox reactors, and its swelling appears to be unaffected by such cycling.

Reference 18 contains an interesting dissertation on uranium swelling which, for the purposes of this discussion, can be said to be dependent largely on the size and distribution of fission-gas bubbles in the metal. In a rather crude sense, it can be said that the swelling of uranium can be held to a minimum by preventing the coalescence of fission-gas bubbles: in other words, although fission-gas generation is a function of irradiation, the effects of such generation can be minimized by preventing the migration and coalescence of the gas bubbles. In even simpler terms the desired condition within the fuel is one in which the number of gas bubbles is kept to a maximum and, consequently, the sizes of the individual bubbles are kept at a minimum. During the production of adjusted uranium, when the material is held in the beta phase, most of the particles of the intermetallic compound UAl_2 and all the particles of U_6Fe precipitated during casting are redissolved. When the material is quenched, the aluminum and iron are retained in solution and, during the subsequent alpha annealing, are reprecipitated as fine particles at the grain

boundaries and within the grains of the material. The quenching and annealing apparently produce a fine intragranular and intergranular dispersion of UAl_2 particles with possibly some iron in solid solution. The reason for the excellent performance of the adjusted uranium is thought to be that this dispersion essentially "anchors" the fission-gas bubbles within the material so that migration and coalescence cannot take place.

Although the adjusted uranium has become the standard material used at the Springfields Works (the fuel-element production facility of the UKAEA),¹⁹ and although the British consider¹⁹ that other alloys now undergoing irradiation are unlikely to show an overall advantage as a fuel material when compared to the adjusted uranium, it is evident that the French

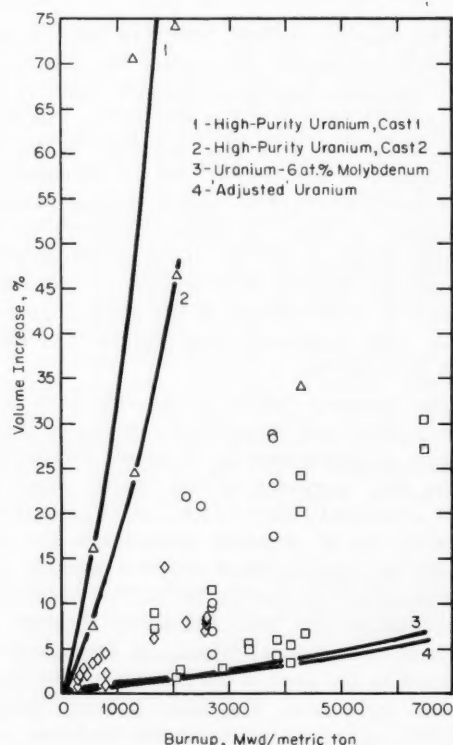


Fig. VII-2 Volume increases in uranium with no added aluminum, irradiated at 500 to 650°C. Curves for high-purity uranium, uranium-6 at.% molybdenum, and "adjusted" uranium are superimposed.¹⁸ \diamond , U. S. results, direct irradiation. \circ , U. S. results, postirradiation annealing. ∇ , AERE results, direct irradiation, 30 to 90 ppm carbon. \square , AERE results, direct irradiation, 300 to 1000 ppm carbon.

consider other alloys as the preferred Magnox-reactor fuel materials. The French flatly contradict the British proposition that the bubble-migration phenomenon is the dominant factor in the growth of uranium fuel elements and state²⁰ that no French experiments have ever shown evidence of such a phenomenon. In the later French reactors, the fuel is typically uranium-molybdenum alloy, with the molybdenum content ranging from 0.5 to 1.0%. These fuel elements are hollow-rod type elements with external cooling only. For their advanced fuel element, which has an annular cross section of uranium cooled internally and externally, the French propose a fuel made of uranium of "low alloy contents" in order to avoid the relatively high neutron absorption of the molybdenum alloys, since the mechanical strength of the molybdenum alloy is not required in the internally cooled annular elements. Perhaps the French philosophy on fuel-material development is that the uranium-molybdenum alloys possess a high mechanical strength at elevated temperatures and that this property makes them particularly suitable for use in the hollow-rod elements subjected to high external pressures. Furthermore, growth becomes negligible at temperatures around 525°C and appears to be independent of the type of alloy considered, although no inconvenient swelling has been observed in the uranium-molybdenum alloy during fuel-element tests.⁶ Although the French are devoting a substantial effort to the development of the annular fuel element that utilizes less absorptive fuel than their typical uranium-molybdenum alloy, they clearly state²¹ that the more conventional closed-tube elements such as those employed in the EDF-2, -3, and -4 reactors will not be abandoned.

Although development of fuel elements for the AGR in Great Britain is continuing, nothing very unusual has been reported concerning the fuel portion of these elements. High-density UO_2 is still the preferred fuel material, and research on this material is proceeding along fairly routine lines. The major emphasis in the development of these fuel elements is in the configuration and cladding materials, which will be discussed in following sections.

Fuel-Element Cladding and Structure

The policy adopted by the British for the development of critical items such as cladding

alloy and structural alloy for fuel elements has been one of parallel development with alternative solutions. It appeared likely that the material used for cladding alloy since 1956 in the Calder Hall reactors would have a limited life because of failure by linkage of grain-boundary cavities at temperatures up to 300°C. This material, a coarse-grained magnesium-0.8% aluminum alloy called Magnox AL80, could be replaced by a fine-grained Magnox AL80 that exhibited a greater ductility and could be expected to have a longer life, but the fine-grained structure could be attained only by the controlled application of cold work which might be expected to be a hindrance to bulk manufacturing operations.¹⁹ The principal alternative to the fine-grained Magnox AL80 alloy was an inherently fine-grained two-phase alloy containing 0.55% zirconium and called Magnox ZR55. Cladding made of bulk materials was produced for the Civil Reactor Program in Great Britain, and large-scale proving trials in the UKAEA reactors were begun. Meanwhile, the manufacture of the fine-grained AL80 cladding material was developed until the process was suitable for large-scale production of cans, and this development turned out to be a fortunate one for the program. Because of the diffusion of plutonium through the Magnox ZR55 cladding at irradiations above 1000 Mwd/ton, the operation of the burst-cartridge detection gear in the Magnox reactors was impaired. Consequently, the fine-grained Magnox AL80 has been adopted as the "standard" cladding material for use in the British reactors.

The requirements of some of the structural components of fuel elements are different than those of the cladding since the loadings are different. For instance, the end caps and associated fittings which support the axial loads acting on stacked elements require a high resistance to creep but demand less ductility than is required of the cladding alloys. To date, a 0.7% manganese alloy, Magnox MN70, has been the one most widely used for end caps and fittings,²² and the most popular material for spreaders and braces has been heat-treated Magnox ZR55 (Ref. 19).

The French presented relatively little information at Geneva concerning the development of cladding and structural materials for the fuel elements of their gas-cooled reactors. Although they are aware of the fine-grained Magnox AL80 used by the British, the French ap-

pear to prefer²³ a magnesium-zirconium alloy similar, presumably, to the Magnox ZR55 considered by the British. Although the typical French fuel element does differ²¹ from the British elements (in that, since EDF-2, the French have fitted their elements with a graphite sleeve), the plutonium diffusion problem must still remain. Little information was given concerning burst-cartridge detection in the French reactors, but the problem has apparently been reconciled with the operation of the reactors. Apparently the magnesium-zirconium alloy is used for the welded end caps²³ of the fuel elements as well as for the cladding material.

The development of cladding materials for higher temperature gas-cooled reactors (such as the AGR and the French gas-cooled heavy-water-moderated reactor EL-4) indicates that, at least for the fairly near future, reactors of this sort will utilize fuel that is clad with very thin stainless steel. Apparently, the British are not considering very seriously any material other than stainless steel for this high-temperature application, although some years ago beryllium had been considered for use in the AGR. The French, however, although resigned to the use of stainless steel at present, continue studies directed toward the development of beryllium as a cladding material and are investigating, in parallel efforts, other materials such as iron-aluminum alloys. Reference 24 contains descriptions of some of the recent work in France on the development of beryllium and indicates that some progress is being made on the problem of improving the ductility of beryllium in the direction perpendicular to the direction of extrusion.

Fuel-Element Configurations

The fuel-element configuration that has been used or specified most frequently in Great Britain is the solid rod in a polyzonal can, whereas the French ordinarily use the hollow rod in a polyzonal can, housed in a graphite sleeve. The British-designed Tokai reactor, however, has²⁵ a hollow-rod element and a graphite sleeve, and the Hunterston reactor also has the graphite sleeve.²⁵ Some of the fuel geometries that have been considered by the British are shown in Fig. VII-3. The arrangement of flat bars in a fuel element is one in which bars with different widths are held par-

allel to each other to "fill" approximately the circular-cross-section fuel channel. The investigation of these geometries has been prompted by the trend to higher heat ratings. A departure from the solid fuel geometry is required if the desired increases in rating are to be achieved. The British have apparently concluded that the flat-bar and tubular elements offer substantial savings in power-generation cost relative to the current solid-rod designs in very large reactors.⁷ Of the two configurations the flat-bar approach is preferred because it appears to have fewer technical uncertainties and slightly better economic potential than the tube.⁷ Of the solid-rod type elements used or specified to date, two types appear to have been used in the largest numbers. These are the polyzonal element with helical fins and the polyzonal element with herringbone fins. Drawings of the external appearance of elements of these types are shown in Fig. VII-4. As is indicated by the drawing, the herringbone element is considerably simpler mechanically than the helical type identified as a Dungeness Mark I element. Although the British have been quite cautious and have stated that the herringbone element has yet to demonstrate its stability in such mechanical form,²⁶ they do plan large-scale irradiation trials and state that the merits of the design²⁷ have re-

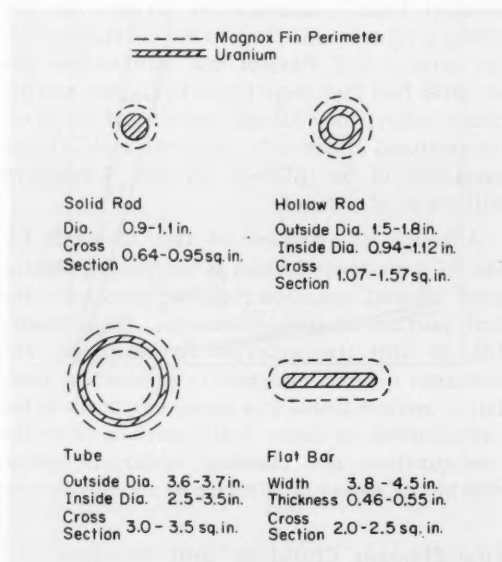


Fig. VII-3 Fuel geometries, British studies.⁷

sulted in its adoption for the second reactor at Sizewell.

The characteristics of French fuel elements for Magnox reactors are shown in Table VII-5, which indicates that the reactors in the EDF series all use hollow rods, that from EDF-2 onward graphite sleeves have been used, and

that for the external surfaces of round elements the herringbone finning arrangement is preferred.

The "advanced" French fuel element that corresponds to the flat-plate type envisioned by the British is the annular element listed in Table VII-5 and shown in Fig. VII-5. As indicated in the drawing, this is a very large element, the distance across the span of the centering device being about 7 in. Its cross-sectional area of fuel and its inside and outside diameters correspond roughly to those studied by the British in their search for an advanced fuel element (Fig. VII-3). It is interesting to compare the maximum specific powers shown in Table VII-5 for the French fuel elements with the data presented in Fig. VII-6, which is based on information generated by the British and presented in Ref. 7. The comparative generation costs shown in the figure are based on optimized reactors of 500-Mw(e) output for each of four fuel geometries. Although it is stated that the fuel prices for the flat bar and the tube are nearly the same for their optimum designs,⁷ the blowing power requirement at a given coolant pressure is lower for the flat-bar design than for the tube, and this more than offsets the costs associated with the requirement for a somewhat larger core when the flat-bar design is used.

With regard to the French fuel-element program, an interesting comparison of the relative costs of the various operations in preparing fuel elements for the reactors for G2 to EDF-3 is shown in Table VII-6. Since the "new uranium" cost should be the same for each ele-

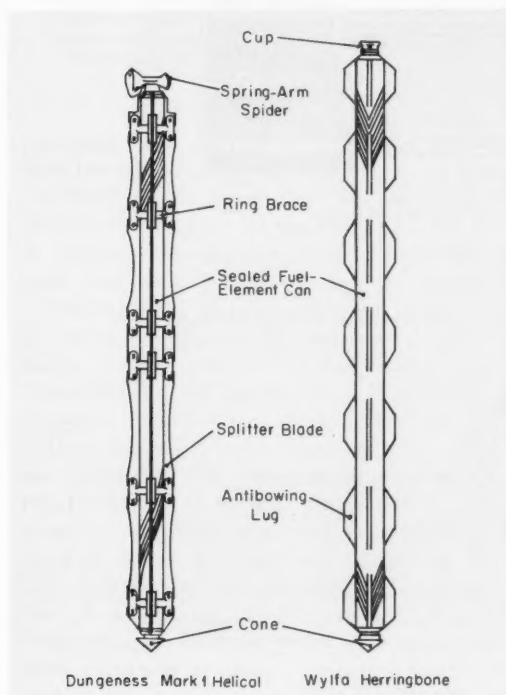


Fig. VII-4 British fuel elements.²⁶

Table VII-5 CHARACTERISTICS OF FRENCH FUEL ELEMENTS²¹

	EDF-1	EDF-2	EDF-3 and EDF-4	Annular element (INCA)
CO ₂ pressure, psia	406	385	435	656
Fuel				
Alloy	U-Mo, 0.5%	U-Mo, 1%	U-Mo, 1%	Slightly alloyed U
Diameter, in.	0.55 × 1.38	0.71 × 1.57	0.90 × 1.69	3.03 × 3.74
Cross section, sq in.	1.24	1.55	1.55	3.88
Maximum specific power, Mw/ton	4.4	5.8	6	11
Can				
Type of fins	Longitudinal	Helical herringbone	Flat-sided herringbone	External, herringbone; internal, longitudinal or corrugated
Maximum heat flux, Btu/(hr)(sq ft)	178,000	260,000	257,000	External, 333,000; internal, 222,000
Element				
Individual support	None	Graphite sleeve	Graphite sleeve	Graphite sleeve
Length, in.	23.6	23.6	23.6	23.6
Number per channel	15	12	15	15

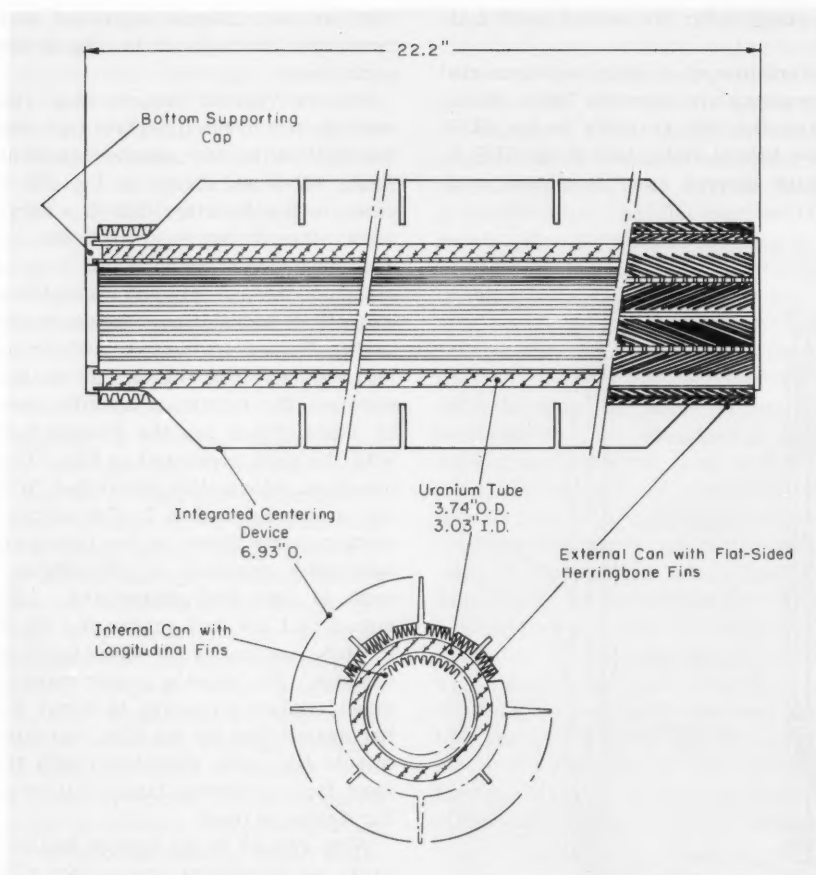


Fig. VII-5 The French annular element (INCA).²¹

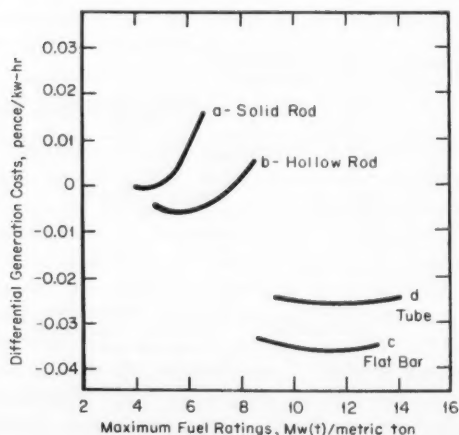


Fig. VII-6 Differential generation costs.⁷

ment type, the relative costs for the complete elements are easily derived.

It appears that no radical changes are forthcoming in the selection of fuel-element configurations for advanced gas-cooled reactors such as the AGR, and the continued use of multirod bundles of small-diameter fuel tubes may be expected.²⁸

Pressure Vessels

The British reactor stations up to Sizewell have utilized steel pressure vessels. The fuel electrical rating has been increased significantly through this series of plants, as indicated in Fig. VII-1, and much of this increase has come about through increases in coolant

Table VII-6 BREAKDOWN OF FUEL CARTRIDGE COST IN FRANCE²³

	Percent of fuel cartridge cost			
	G2 and G3 solid rod (dia., 1.22 in.)	EDF-1 hollow rod (dia., 0.55 to 1.38 in.)	EDF-2 hollow rod (dia., 0.71 to 1.57 in.)	EDF-3 hollow rod (dia., 0.90 to 1.69 in.)
New uranium (ingots)	72	58	56	54
Solid-rod and hollow- rod manufacture	6	7	6	7
Cans and plugs	4	7	4	5
Alumina pellets		1		
Centering grids			2	
Graphite sleeves			8	8
Manufacture and testing	18	27	24	28

pressure. But increases in design pressure have been limited by the difficulties met in procurement and in on-site welding and inspection of thick steel plate.⁷ At the same time the trend in reactor capacity has been to high-capacity units that make possible significant reductions in capital cost, and the influence of this trend has been to increase reactor size despite the higher power density that has been achieved. Consequently the problem of increasing the pressure in order to increase the fuel rating, while at the same time larger and larger plants are achieved, has resulted in the conflicting requirements of higher pressures in larger vessels. An example of this is evident in the designs of the Bradwell and Dungeness reactors. For Bradwell, 3-in.-thick plate was specified at a working pressure of 147 psia. For Dungeness, about three years later, when 4-in.-thick plate was available, coolant pressure was limited to a value somewhat less than 300 psia,

and pressure-vessel dimensions were at a maximum of approximately 60 ft in diameter. The use of prestressed-concrete pressure vessels has permitted the economic optimization of coolant pressure and vessel diameter, and it is obvious from inspection of Fig. VII-1 that the desired pressure of the larger and newer power plants is substantially greater than that which had been possible in the units up through Dungeness-A.

The impressive sizes of the concrete pressure vessels which have been specified for planned plants are illustrated by the data presented in Table VII-7. For example, the Wylfa pressure vessel has an internal diameter about half again as great as that of the largest steel vessels, and the nominal pressure of its primary coolant is more than a third again as great as that of the steel-vessel plants. The proponents of the concrete vessel claim that it has other advantages also. A fair body of opin-

Table VII-7 GAS-COOLED CLAD-FUEL REACTORS USING CONCRETE VESSELS^{4,10}

Reactor	Date of first reactor on power	Thermal power, Mw	Primary coolant nominal pressure, psia	Primary coolant temperature at reactor outlet, °F	Pressure vessel		
					Shape of internal cavity	Dimensions of internal cavity, ft	Approximate minimum wall thickness, ft
G2	1959	250	215	644	Horizontal-axis cylinder	Diameter, 45.9; length, 51.5	9.8
G3	1960	250	215	644	Horizontal-axis cylinder	Diameter, 45.9; length, 51.5	9.8
EDF-3	1966	1560	435	770	Vertical-axis cylinder with truncated cone sections at top and bottom	Diameter, 62.3; height, 69.1	13.9
Oldbury	1966	890	365	770	Vertical-axis cylinder	Diameter, 77.0; height, 60.0	15.0
EDF-4	1968	1650	435	752	Vertical-axis cylinder with upper trun- cated cone section	Diameter, 62.3; height, 119.0	16.4
Wylfa	1968	1870	400	770	Sphere	Diameter, 96.0	11.0

ion among the British and French designers is that the steel pressure vessels used in the United States are, in fact, susceptible to sudden failure. In the opinion of these people, the maximum credible accident should logically be considered to be the sudden brittle fracture of the reactor pressure vessel; the mode of ultimate failure of concrete pressure vessels, however, is progressive and without sudden rupture.²⁹ Obviously this theory is not widely popular among the proponents of steel vessels.

Refueling Machines

The performance of refueling equipment in both the British and French reactors has been acceptable, and the development of refueling machines has been characterized by steady improvements based on operating experience gained with the machines used in the relatively early reactors. In the G2 and G3 reactors, the original machinery has been improved principally for increasing operational safety, but the improvements have also permitted increases in the refueling rates so that it is now possible to discharge nearly 27 fuel channels in 24 hr, or about three times the initial design rate of the machines.¹ The EDF-2 reactor is equipped with two refueling machines, the main machine and an auxiliary machine, whereas in EDF-3 the French have reverted to a single all-purpose device. Among the difficulties encountered by the designers of the equipment for these reactors is the problem of lifting of the last few elements in a particular channel when that channel is being unloaded under power. Important design features of the machines³⁰ are the baffling arrangements that are used to avoid levitation of the final elements by preventing a very high loss of head at the top end of the channel being refueled. The performance of the French refueling equipment will be interesting to observe in view of the fact that studies and tests are said to account for about 35% of the cost of the machines.

A British paper³¹ relates the experience obtained with the Bradwell and Latina on-load fuel-handling equipment and discusses the influence of this experience on the design of later machines for the Dungeness and Oldbury reactors. It is sufficient here to point out two of the major conclusions of the paper. The first is that the time required for the machine to per-

form a certain number of reloading or charging operations is considerably longer in practice than is indicated by design calculations. These differences in time are illustrated by the data in Table VII-8 which show that the actual time is some $2\frac{1}{2}$ times as great as the design time. It is felt that the required time may be reduced to a value of about twice the design time. The other conclusion drawn by the British is given in the following quotation from Ref. 31: "Massive rather than elegant devices give by far the best service."

Table VII-8 BRADWELL CHARGE-MACHINE OPERATING-CYCLE TIMES³¹

Operation	Actual time, hr	Design time, hr
Inspect three channels of fuel and load magazine	$1\frac{1}{2}$	
Load three channels of fuel into machine	2	1
Travel machine to gas supply and connect	$\frac{1}{2}$ }	$\frac{1}{4}$
Pressurize machine	$\frac{3}{4}$ }	
Reinstate previous standpipe and prepare new one	3	
Connect machine to standpipe	$\frac{1}{2}$	$\frac{1}{2}$
Discharge and reload three channels	8	$3\frac{3}{4}$
Disconnect machine and travel to vent point	$\frac{1}{2}$	$\frac{1}{4}$
Depressurize machine	$\frac{1}{2}$ }	2
Connect to discharge hole and unload three channels	2 }	
Travel to vent point and purge machine	$\frac{1}{2}$	
Travel to loading point and connect	$\frac{1}{2}$	$\frac{1}{4}$
Total times	$20\frac{1}{4}$	8

Control and Instrumentation

The development of larger and more complex reactor plants has had a number of effects on plant operating philosophy. There are trends (1) toward centralized control,³² (2) toward a better understanding of, and consequently a closer approach to, the temperature limits of the various core materials,³³ and (3), because of the very large reactors currently under construction and operation, toward concentrating on the control and stability problems inherent in very large cores, particularly with regard to spatial variations.³⁴ The experience gained

through the years, while revealing some problems that had not been anticipated in the early designs, has served as both a firm foundation and an experimental basis for the future operation of power reactors.³⁵

At the time the burst-cartridge detection gear was designed for the Calder Hall reactors, no operating experience existed to indicate the types of fuel-element failure which might occur. Consequently the systems supplied for the early reactors were designed to meet a few fundamental requirements for burst-cartridge detection and were not particularly suited to detecting properly the types of actual bursts that would be encountered during operation. As a consequence of reactor operation and the determination of the most likely modes of fuel-element failure, system specifications have been more clearly defined. The fast burst, a fuel-element failure that gives virtually instantaneous release of a significant quantity of fission products, has been identified as a particular failure type, and this type of burst makes important the quick-detection-and-location capability that has led to an increase in the frequency of group scanning at some expense to sensitivity. Since experience with the early reactors became available, the burst-cartridge detection systems for the British reactors have used a combination of the fast scan at low sensitivity with a single-channel scan of long cycle time. Some of the features

of systems designed for later British reactors are given in Table VII-9.

The French G2 reactor was initially equipped with a burst-cartridge detection installation similar to that used in the Calder Hall reactors.¹ On the strength of experience acquired with the G1 reactor, modifications to the G2 system and development of the G3 system have been based on a capability which the French feel is important: measuring the time rate of the signal as well as an instantaneous absolute value. The G3 reactor installation has performed satisfactorily and has served as the basis for the burst-cartridge detection systems that will be employed in the EDF reactors.

Advanced Concepts and Future Directions

The concepts that are classified here as "advanced" include the advanced gas-cooled reactor, typified by the Windscale AGR, and the integrated plant designs that utilize concrete pressure vessels to enclose the entire primary system.

The AGR has been described rather thoroughly in the literature, and here it will suffice to say that the reactor was commissioned essentially on schedule and that its operation since commissioning has been entirely satisfactory.

Table VII-9 DETAILS OF RECENT BRITISH BURST-CARTRIDGE DETECTION SYSTEMS³⁶

	Dungeness	Sizewell	Oldbury	Wylfa
Reactor No. 1 on load	1964	1965	1966	1968
Channels in reactor	3932	3784	3308	6156
Channels per group sample	32	32	8	32
Period of group scan, min	6.7	Continuous	11	Continuous
Cycle time per sample, sec	25	60	25	60
Single-channel scan period, hr	3.7	4.25	4.0	7.1
Total precipitator units per reactor	20	18*	27	29*
Group sample flow rate at precipitator, lb/sec	0.115	0.037	0.115	0.031
Single sample flow rate at precipitator, lb/sec	0.042	0.017	0.042	0.026
Total circuit mass flow, lb/sec	1.6	5.2	2.4	7.3
Circuit pressure drop, psi	40-45	40	42-47	55
Transit time, channel to precipitator, sec	6	5.9	11	6.7
Sensitivity (M.D.A.)†				
Group, mm ²	10.0	3.5	2.5	3.8
Single, mm ²	0.57	0.11	0.57	0.13

*Eight chambers per precipitator unit.

†Minimum detectable area as determined by an alarm set 25% above background for a scanned system and 15% above background for a continuous system.

As mentioned briefly before, the AGR type fuel elements have undergone no major evolution in the past few years: the reactor type is apparently destined to use the multirod bundle type element. Since an element of this type involves relatively high fabrication costs per unit weight of uranium, favorable economic performance requires a rather long fuel life in the reactor. This requirement poses a difficult task for the designer if it is also imperative to design for the minimum possible fuel enrichment; for, in the absence of a good low-cross-section material for the fuel jacket, he must design for the minimum-thickness jacket of stainless steel. A fairly recent development in the AGR fuel program is the consideration of the use of plutonium³⁷ as the enriched component of the fuel, since it is felt in Great Britain that the stockpile of plutonium is growing at a rate which will ensure a more than adequate supply of the material when fast-breeder reactors become commercial. Consequently the British philosophy appears to be that the thermal reactors will be permitted to utilize this "extra" plutonium if it appears feasible.

The integrated plants, which are EDF-4, Oldbury, and Wylfa, are typified by their compact arrangements (if such a description can be applied to these huge reactors). Although all three reactor vessels contain all the principal components of the primary circuit, their shapes and arrangements are quite different, as indicated in Table VII-7. EDF-4, for example, utilizes a rather tall, thin configuration in which the steam generator is located below the reactor core. A disadvantage of this scheme is the lack of natural-circulation cooling in the event all CO₂ circulators are lost. With the over-and-under arrangement of the major components of the primary loop, it follows that the EDF-4 pressure-vessel internal space is relatively small in diameter and quite high, as indicated in Table VII-7. The basic arrangements in Oldbury and Wylfa resemble one another in that the steam generators are located peripherally around the reactor core and at about the same elevation as the core. The shapes of the internal cavities of the pressure vessels are different, however, with the Wylfa station retaining the traditional spherical shape of the earlier British stations and with Oldbury using a vertical-axis cylinder.

The directions of future development of the planned stations in both Great Britain and

France are rather clearly shown by the information presented in Table VII-1 and by the increases in primary coolant pressure and fuel electrical rating shown in Fig. VII-1. What is not so clearly shown is the type reactor that will be employed for the "next generation" plants in these two countries.

At the time of its conception, the AGR was felt to be the logical successor to the Magnox stations in Great Britain. Although the AGR has performed as expected, there is still the problem of minimizing absorption in the fuel cladding in order to obtain long fuel lifetimes at low values of enrichment. When one makes the jump to enriched fuel, a variety of reactors enters the picture. Furthermore, although the AGR fuel has behaved well to date, the operation reported has carried the exposure only to the neighborhood of 4000 Mwd per metric ton of uranium, a long way from the economic optimum average burnup of around 20,000 Mwd per metric ton of uranium, or even the average value of 12,000 Mwd per metric ton of uranium which is stated²⁸ to be the target for the first charge of an early commercial station. Effort is being directed toward the further development of the AGR fuel element along such lines as thinner cladding material, basic studies in oxide fuels, and the development of plutonium enrichment for thermal-reactor fuel. At this time the preferred future concept for construction in Great Britain is not clear, although there have been some indications that the Magnox reactor has about reached the end of the line. Whether the AGR type plant will be built in large numbers as a second-generation gas-cooled-reactor type is uncertain. The selection of a preferred concept for the future will be based largely on economic considerations, but surely there will be a strong tendency toward the gas-cooled type because of the large amount of experience accumulated and the advances expected of the very large integrated plants utilizing concrete pressure vessels.

In France the directions of future development are clearer: the French like Magnox reactors, and, after some apparent early uncertainty, they have committed their gas-cooled program to the concrete pressure vessel. They are developing the integral plant concept with the EDF-4, and they will apparently adopt no other new concept in the very near future, although the performance of EL-4 will affect later plans. In France, then, it appears that the Mag-

nox reactor can look forward to a relatively bright future.

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Section VIII

Power Reactor Technology

Gas-Cooled Unclad-Fuel Reactors

The gas-cooled unclad-fuel reactors are characterized by high coolant temperatures and all-ceramic fuels. The two reactors of this type which are under construction, the Dragon experiment in England¹ and the Peach Bottom prototype in the United States,² have not changed markedly since their designs were described in past issues of *Power Reactor Technology*. Dragon was described in the December 1962 issue, 6(1): 74, and the Peach Bottom description appeared in the June 1962 issue, 5(3): 61. The Dragon reactor³ achieved initial criticality on Aug. 23, 1964, and initial operation of the Peach Bottom plant⁴ is expected in 1965.

The designation of Dragon as an "experiment" and of Peach Bottom as a "prototype" is explained by the difference in components provided for the two plants. The Dragon installation is not equipped with power-generating equipment—its heat is simply dumped to a heat sink—whereas the Peach Bottom plant is pro-

vided with the usual electric-power-generating components of a power plant. A comparison of the characteristics of the two reactors is given in Table VIII-1, which contains information taken largely from Refs. 4 and 5.

As seen in Table VIII-1, the coolant outlet temperatures for two reactors are identical. From an overall point of view, this is perhaps the most significant comparison that can be made between the two reactors since both use the same coolant gas and both have the attainment of high coolant temperature as a major design objective.

As indicated in Table VIII-1, the Dragon core will be loaded with two types of fuel elements. Since the core is quite small, its neutron leakage is very high, and it does not, as a whole, represent a core typical of larger power reactors. It uses a two-zone loading⁶ to overcome this disadvantage and generate information applicable to larger cores. The seven center

Table VIII-1 CHARACTERISTICS OF DRAGON AND PEACH BOTTOM

Plant characteristics	Dragon	Peach Bottom
Reactor thermal power, Mw	20	115
Net electric power, Mw	None	40
Coolant gas	Helium	Helium
Pressure, psia	294	350
Inlet temperature, °F	662	652
Outlet temperature, °F	1382	1382
Reactor core	Vertical cylinder	Vertical cylinder
Height, ft	5.25	7.5
Diameter, ft	3.5	9.1
Average power density, kw/liter	14.0	8.3
Fuel elements	Seven-rod clusters	Individual tubes
Number	37	804
Fuel material	Uranium in driver elements; uranium and thorium in test elements	Uranium and thorium
Cladding material	Graphite	Graphite
Control elements	Rods	Rods
Number	24	36
Location	Radial reflector	Core
Poison material	B ₄ C	B ₄ C
Drive location	Top	Bottom
Refueling	At shutdown	At shutdown

elements of the core will be representative of those which would be used in a large power reactor and will be fueled with a uranium-thorium mixture. The remaining 30 elements, which surround the seven center elements, will contain no thorium and will serve as driver elements that must be replaced at relatively frequent intervals. By this means the center elements can be irradiated for several years and will effectively "see" conditions that would be typical of a large power-reactor core.

Materials developments for the reactor type in the United States and the United Kingdom have proceeded along quite similar paths. Graphites with helium permeabilities as low as about 10^{-3} cm²/sec have been developed in both countries, and fuel-particle development has been directed principally toward the design of coated particles.^{7,8} Particle-coating investigations are concerned with single, double, and triple coatings utilizing pyrolytic carbon and silicon carbide. In addition to the work on pyrolytic carbon and silicon carbide, work continues in both countries on the development of alumina and beryllia.

The well-known advantages of helium as a reactor coolant are restated in the following quotation from Ref. 2:

Helium gas is used as the HTGR system's coolant because its chemical inertness and other properties contribute to reliable, easily maintainable, high temperature plant operation. Helium does not react with or corrode any structural material; it does not react with steam, water or air. It does not undergo a change in phase at operating temperatures, nor does it solidify at ambient temperature. Helium does not parasitically absorb in the reactor core, and cannot affect the physics of the reactor; it does not become radioactive . . .

There are, however, disadvantages to the use of helium, and these center about the fact that it is a relatively high-priced gas. The pressure of the system must be relatively high in order to achieve good heat transfer and heat transport with reasonable pumping powers. High pressure, unfortunately, tends to aggravate the leakage problem that exists with helium. Leakage must be minimized because of cost considerations and, in reactors of the unclad-fuel type, because of the relatively high radioactivity of the primary coolant. Consequently, the ability to fabricate primary coolant systems with a very high standard of overall leaktightness has been of major importance in the Dragon and Peach Bottom plants. Recent tests of the Dragon

facility⁹ have been very encouraging in this respect since the actual leakage rates encountered were within the specified rate of 0.1% per day of the gas contained in the circuit.

Although technologies developed for Dragon and Peach Bottom have been extended to conceptual designs of large nuclear power plants, the actual directions of future development of plants of this type are not uniquely defined at this time. The developers of Peach Bottom consider a plant in which a fuel-element purge system is employed and fission-product release is encouraged in order to obtain conversion ratios greater than unity.² Actually, this is the approach that was used by the Dragon designers⁵ in the early stages of the project—they envisaged a fuel element that emitted as many fission products as possible to the fission-product trapping system, thereby obtaining the maximum reactivity benefit through the removal of neutron poisons from the fuel. However, further work showed that some of the more important fission products tended to remain in the fuel, although enough did leave to require the use of a complicated purification system to limit primary coolant activity. Consequently, the Dragon project does have the purge and purification system, but the hoped-for reactivity benefits from the removal of neutron poisons have not materialized. Recent studies, which must be regarded as preliminary, indicate that there could be some cost benefit associated with the development of a fuel element which could be directly cooled by the main helium stream and which would not be supplied with the elaborate purging and purification systems currently employed in Dragon and Peach Bottom.

The pebble-bed reactor¹⁰ represents a different version of the high-temperature gas-cooled concept. It also uses the fuel in completely ceramic form—probably in balls containing UC₂-ThC₂ particles coated with pyrolytic graphite, with a graphite shell surrounding each ball—but no attempt is made to draw off fission products. The reactor is fueled more or less continuously during power operation by adding balls to the top of the assemblage, which constitutes the active portion of the core, and draining off balls from the bottom. Of the balls removed, the most highly exposed and those which may be damaged are discarded, and the good balls of low exposure are recycled to the reactor. One of the developmental prob-

lems is to find a means of handling and sorting the balls.

The concept is being developed primarily in Germany, and an experimental power station of 15 Mw(e) (the AVR Reactor), built for a group of utility companies with financial support from the federal government, is reported to be near completion in the Federal Republic of Germany.¹⁰ Development is also proceeding toward a large reactor of the type.

It would appear that the development of the high-temperature gas-cooled reactor, utilizing unclad fuel, must depend heavily on the experimental information and the operating experience gained from the plants now under construction or startup. Until this experience has been gained, speculation as to the future development of these already sophisticated designs should be considered as just that.

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Section

IX

Power Reactor Technology

Sodium-Cooled Reactors

By J. R. Dietrich

Technology of Sodium Systems

The two currently important types of sodium-cooled reactors are the sodium graphite reactor and the fast breeder. They differ markedly from each other in some characteristics, and, moreover, there has been increasing interest during the past two or three years in fast breeders using other coolants. Nevertheless, the technology of sodium cooling is important enough to both types to justify discussing them together and in terms of the developments in sodium systems.

The Conference produced reviews of heat transfer^{1,2} and corrosion^{3,4} in liquid-metal systems, by both U. S. and Russian authors, of value to the designer of sodium-cooled reactors. The heat-transfer papers, which include such practical considerations as burnout, and heat transfer in rod bundles, will be particularly useful. The discussion here, however, will be confined to those aspects of sodium technology which affect the direction of reactor development in more obvious, qualitative ways.

Fission Products in Sodium

One of the important areas of new information is in the behavior of fission products in sodium. Investigations by Atomic International^{5,6} have shown that liquid sodium will trap radioiodine quantitatively, and this characteristic is expected to be of considerable significance in determining siting criteria for sodium-cooled reactors. The Dounreay Fast Reactor (DFR) has utilized vented fuel elements, of a uranium-20 at.% molybdenum alloy, which purposely allow the escape of fission products into the sodium coolant.⁷ Russian experience with

the BR-5 reactor⁸ has given other information on the behavior of fission products in the primary coolant system which is worthy of note. A brief summary follows.

The BR-5 reactor, which was fueled with plutonium oxide elements in stainless-steel jackets, began operation in 1959 and reached the design value of fuel burnup (2%) the following year. After a burnup of 2.4% it was noted that the residual activity of the sodium, after the decay of ²⁴Na (half-life, 15 hr), began to increase with fuel exposure. Previously the residual activity had all been attributable to the small amount of ²²Na (half-life, 2.6 years) formed by the (*n*,2*n*) reaction on ²³Na. Most of the new activity was due to ¹³⁷Cs. Table IX-1 shows the growth of this activity, relative to the ²²Na activity, with fuel burnup, and this growth is plotted in Fig. IX-1 in terms of the total residual activity. During the period after the ¹³⁷Cs became apparent, ¹³³Xe was detected in the argon-gas blanket over the sodium. By the time the fuel burnup had reached 5%, alpha activity due to plutonium was found in the sodium (Table IX-1).

After 5% burnup the reactor was shut down because of trouble with a circulation pump, and all the fuel was removed. Of the total of 81 fuel assemblies, 18 were found to contain fuel elements showing gross leakage of fission products. After decay of ²⁴Na the sodium was drained from the primary system, but the drainage caused no substantial reduction in the activity of the primary system, indicating that the fission products were deposited on pipe walls and other surfaces in the system. The primary system was decontaminated by first washing with steam and then with solutions of 0.5% KMnO₄, 5% nitric acid plus 1% oxalic acid, and finally with distilled water. This treatment reduced the gamma background in the primary system cells from several thousand micro-

Table IX-1 GROWTH OF ^{137}Cs ACTIVITY IN BR-5 PRIMARY COOLANT,⁵ RELATIVE TO THAT OF ^{22}Na

Fuel burnup, %	Residual activity induced by ^{22}Na , %	Residual activity induced by ^{137}Cs , %	Alpha activity (dis/min per gram of Na), %
2.00	100		
3.20	80	20	
3.90	60	40	
4.55	30	70	
5.00	5	95	50

roentgens per second to about $10\ \mu\text{r}/\text{sec}$. The major activities in the decontamination solutions were ^{137}Cs , ^{95}Zr , and ^{95}Nb . The reactor was reloaded with 63 of the original plutonium oxide fuel assemblies (some of which apparently contained leaky elements) and 50 new assemblies containing UO_2 elements. A bypass circuit was installed in the primary line which could be valved off at will to allow decay of ^{24}Na in the bypass line without interrupting operation of the reactor. A program of experiments on

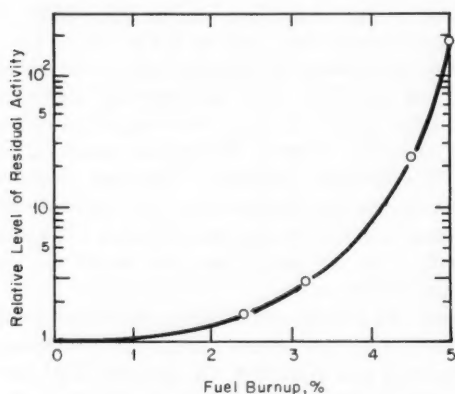


Fig. IX-1 Growth of the residual activity of the primary circuit of BR-5.⁶

the behavior of fission products produced the following results:

1. With the reactor operating at 100 kw with a sodium outlet temperature of 350°C , 90% of the residual gamma activity in the coolant was due to ^{137}Cs , 5% to ^{22}Na , and the remainder to ^{140}Ba , ^{95}Zr , ^{95}Nb , and ^{140}La . The cesium activity level was 4×10^5 dis/min per gram of sodium. Draining of sodium from the bypass line decreased the activity of the line by only 10 to 20%.

2. Changes in sodium temperature in the primary circuit and in sodium flow rate through the bypass section did not produce a noticeable effect on the concentration of cesium on the inner walls of the circuit.

3. Heating of the walls of the bypass section, while the coolant was circulating, decreased the concentration of ^{137}Cs on the inner walls by several times.

4. With the reactor shut down, after the decay of ^{24}Na , and with the oxide cold traps shut down, heating of the walls of the entire primary circuit caused the ^{137}Cs concentration in the coolant to increase by several times. After the cold traps were put into operation, the ^{137}Cs concentration quickly diminished to the initial level. In the steady state the cesium content of the coolant depends on the cold-trap operating conditions; an increase of cold-trap temperature from 110 to 185°C caused the ^{137}Cs activity in the coolant to increase by a factor of 3.

5. Items of equipment, such as pumps and level gauges, which had some parts submerged in the coolant and some parts in the cover gas, were found to be contaminated by ^{137}Cs in both regions. The contamination was a maximum in the region of the interface between sodium and cover gas.

6. The argon cover gas was found to contain the following radioactive isotopes in addition to ^{41}Ar : ^{133}Xe , ^{129m}Xe , ^{135}Xe , ^{135}Cs , ^{85}Kr , ^{85}Rb , and ^{88}Rb . The important contributors to the activity are ^{133}Xe and ^{135}Xe . The major components of the activity from a sampling during reactor operation at 1000 kw were as follows:

^{133}Xe	$4.3\ \mu\text{C}/\text{cm}^3$
^{135}Xe	$0.95\ \mu\text{C}/\text{cm}^3$
^{133m}Xe	$0.05\ \mu\text{C}/\text{cm}^3$
^{85}Kr	$0.017\ \mu\text{C}/\text{cm}^3$

The reference calls attention to the fact that no fission products with half-lives less than 2.5 hr were found in the cover gas, although the equipment was capable of detecting half-lives as short as 5 min.

Sodium Handling and Cleanup

The experience in the general handling, cleanup, and maintenance of quality of sodium has been much greater than the amount of reactor operation would indicate, and this phase of sodium technology appears now to be well understood. This was not the case, however,

when the DFR was first commissioned: considerable trouble was experienced in the initial cleanup of the system.³³ References 5 and 9 describe the filling and cleanup of the systems for the Hallam Nuclear Power Facility (HNPF), the Experimental Breeder Reactor (EBR-II), and the Enrico Fermi Fast Breeder Reactor (EFFBR). The Russian experience in reloading the fuel elements of BR-5 after examination is worth noting.⁸ The residual sodium and sodium oxide were not removed from the fuel assemblies. The assemblies were reloaded, a few at a time, into the reactor sodium system, and the oxide was cleaned up by the normal system cold traps. The cleanup of carbon from the sodium of the Sodium Reactor Experiment (SRE) after it had been contaminated by the inleakage of Tetralin, as described in Ref. 5, is also of interest.

Despite the routine solution of sodium cleanup problems, it is still true that the protection of sodium systems from the intrusion of oxygen and other contaminants is a consideration of primary importance. The removal or replacement of components in sodium systems is always done with great care to preserve the cleanliness of the sodium system, and, even when the sodium is drained, care is taken to maintain the inert atmosphere of the cover gas throughout the system. Thus, when it was necessary to perform fairly extensive repairs to the hold-down mechanism of the EFFBR, these repairs were made by workmen who entered the drained reactor vessel through an air lock and worked in gastight suits. The scheme that has been used for removing a primary sodium pump from the EBR-II reactor, without contaminating the cover gas, is illustrated in Fig. IX-2. A system similar in principle has been used on the EFFBR for the removal and replacement of components.

Manipulations Under Sodium

Experience has continued to affirm that the manipulation of components under sodium—and particularly the handling of fuel elements in fast reactors, where rather high precision is required of the mechanisms—is one of the more difficult aspects of sodium technology. It has been pointed out frequently that the opacity of the sodium, and the resulting necessity of carrying out blind operations, adds considerably to this problem. Experience with the EFFBR⁹

has shown that relatively minor malfunctions may lead to more serious results because of the difficulty of detecting them. Thus the offset handling mechanism was bent in the course of the preoperational test program when an attempt was made to move it laterally with a partially raised dummy subassembly still in the gripper. Damage also occurred to the hold-down mechanism during this program and appears to have been increased by the continued use of the mechanism after initial slight (and undetected) damage caused by the sticking of a dummy subassembly to one of the fingers of the mecha-

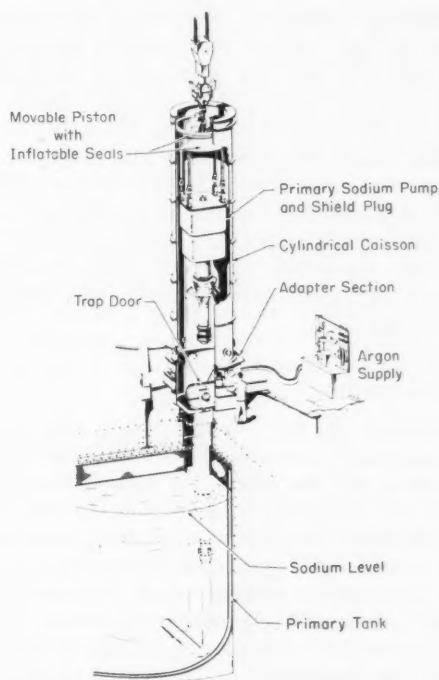


Fig. IX-2 Apparatus used to remove the EBR-II sodium pump.⁹ The adapter section, which includes the gastight trap door, was installed, with a gastight seal, around the shield plug of the pump. The cylindrical steel caisson was sealed to the top of the adapter, was purged and filled with argon. The shield plug, with the pump attached, was then lifted slowly by the crane, while the caisson was kept gastight by the movable piston and argon was admitted to hold the pressure constant. After the pump cleared the trap door, the door was closed, resealing the reactor cover-gas system, and it was no longer necessary to maintain the argon atmosphere in the caisson. The operation was carried out with the reactor sodium temperature at 260°F. After the pump had been withdrawn and cooled to room temperature, it was transported to the cleaning and repair area inside the caisson.

nism. A "sweep arm" has been installed to prevent the recurrence of the latter difficulty. This sweep arm can be swept in a horizontal arc to determine that no core component is attached to the hold-down mechanism before lateral motion of the mechanism is initiated.

Preoperational tests of the fuel-handling equipment of EBR-II indicated the desirability of incorporating means of manual operation of the fuel-transfer arm to permit the operator to feel any interference that might be present.⁸ This was accomplished by a system of counterweights to balance the weight of the rather heavy mechanism.

Although problems have been encountered with the fast reactor fuel-handling systems, it appears that they are problems of arriving at satisfactory designs for specific pieces of equipment rather than general feasibility problems. Indeed, the experience with practice handling of the fuel in both EBR-II and the EFFBR indicates that fuel handling is satisfactory once the specific troubles have been corrected. The British experience with fuel-element handling in the DFR appears to have been good from the beginning.⁷

Components

Outside of the reactor itself, the major components¹⁰ common to sodium-cooled power reactors are the pumps, the steam generators, and the intermediate sodium-to-sodium heat exchangers. The use of the latter, along with the second set of pumps and piping required for the intermediate sodium circuit, must be reckoned as one of the economic handicaps of the sodium-cooled reactor. Apparently it will not soon be eliminated, for no recent design has appeared without the intermediate sodium system to isolate the radioactive primary sodium from the water in the steam generator. Technically, no serious problems have been found in the design of the intermediate heat exchanger, although the units built to date do not approach the capacities that will be required for plants of economic size and do not reach the temperatures that are ultimately hoped for. The three exchangers for the EFFBR plant, of 143 Mw(t) capacity each, are the largest so far. They operate with primary inlet and outlet temperatures of 900 and 600°F, respectively, and secondary inlet and outlet temperatures of 520 and 820°F.

The steam generators designed to date show great variety of concept. The HNPF and EBR-II units are natural-circulation boilers that use double-wall tubes. In the former the sodium is on the tube side, whereas in the latter it is on the shell side. The EFFBR unit is a once-through generator with the water in single-wall tubes and sodium on the shell side. The three HNPF generators, each of which has a capacity of 69 Mw(t) in the evaporator and 16 Mw(t) in the superheater to produce steam at 895 psig and 875°F, are the only ones that have operated at power. No difficulties have been experienced other than the necessity of controlling the water level very precisely (within ± 1 in.) to give satisfactory operation without excessive carry-over of water. The design, which utilizes re-entrant tubes to circumvent thermal-expansion problems, has been described previously in *Power Reactor Technology*, 5(3): 44-47.

The once-through design of the three EFFBR steam generators is shown in Fig. IX-3. Before operation the tubes of one of these units were found to be cracked at the bends as a result of stress corrosion. The corrosion was attributed to residual cleaning solution containing sodium hydroxide and sodium nitrate. During isothermal operation, after about two weeks of test with maximum sodium flow, several tubes

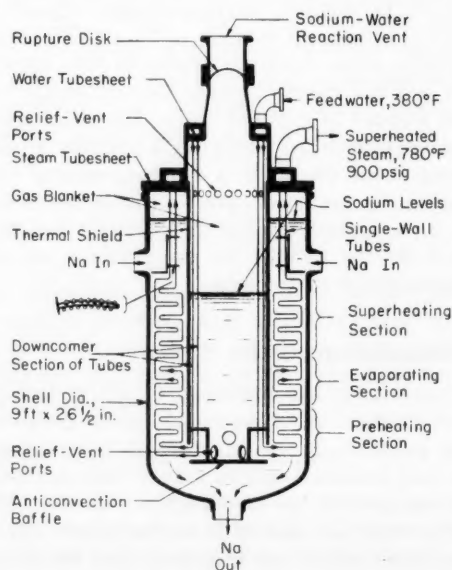


Fig. IX-3 Diagram of the EFFBR steam generator [143 Mw(t)].¹⁰

failed in another of the units. The resulting sodium-water reaction generated enough hydrogen to break the rupture disk (see Fig. IX-3), which was set at 60 psi. All systems designed to accommodate such a reaction functioned satisfactorily, and the separator collected all the sodium expelled from the unit. This inadvertent demonstration of the ability to accommodate a large sodium-water reaction is considered an important contribution to steam-generator development.¹⁰ The tube failure was attributed to flow-induced vibration of the tubes opposite the sodium inlets. Baffles and tube clips have been installed to correct the trouble.

It is stated¹¹ in the description of the proposed BN-350 fast-breeder reactor in the USSR that the steam generator will be of the natural-circulation type, with a single separating wall between sodium and water. The choice of single-wall construction is said to have been substantiated by a number of experiments on wall ruptures, and sodium-water and sodium-steam interactions, in models of steam generators.

In the studies of large advanced sodium graphite plants,⁵ there has evolved the concept of steam-generator systems made up of many low-capacity modules. Separate modules are proposed for the evaporator, superheater, and

reheater functions, each with a capacity of 12 to 15 Mw(t). The superheat and reheat modules would be of stainless steel, and the evaporators would be of chrome-molybdenum steel. All modules are of vertical, once-through design, with single-wall tubes and with the sodium on the shell side. The problems of caustic and chloride stress corrosion are minimized by avoiding crevices and by the use of ferritic steels for the evaporator modules. The justification for the use of single-wall tubes is the choice of shell thickness adequate to contain full steam pressure for short periods of time, plus the provision of rupturable membranes, spaced axially along the shell length, to rupture below full steam pressure and direct any reaction products resulting from a tube break to a relief tank. The modular array is expected to allow the isolation and removal from service of any unit that may give trouble, with a minimum effect on plant availability.

The most definite development in sodium components seems to be the virtual abandonment of electromagnetic pumps in favor of mechanically driven centrifugal pumps. The characteristics of the pumps for HNPF, EBR-II, and EFFBR are listed in Table IX-2, and drawings of the primary pumps for the three

Table IX-2 DESIGN CHARACTERISTICS OF SODIUM PUMPS FOR HNPF, EBR-II, AND EFFBR

	HNPF	EBR-II	EFFBR
Primary system pumps			
Type	Mechanical free surface	Mechanical free surface	Mechanical free surface
No. of units	3	2	3
Capacity, gal/min	7200	5500	11,800
Dynamic head, ft	160	200	310
Design temperature, °F	1000	800	1000
Motor speed, rpm	900	1075	900
Motor power, hp	350	350	1060
Sealing arrangement	Mechanical shaft seal	Totally enclosed drive motor	Mechanical shaft seal
Material	304 S.S.	304 S.S.	304 S.S.
Type of speed control	Eddy-current coupling	Variable frequency and voltage	Wound rotor motor with liquid rheostat
Manufacturer	Byron-Jackson	Byron-Jackson	Byron-Jackson
Secondary system pumps			
Type	Mechanical free surface	A-c linear induction	Mechanical free surface
No. of units	3	1	3
Capacity, gal/min	7200	6500	13,000
Dynamic head, ft	170	142	100
Design temperature, °F	1000	700	1000
Motor speed, rpm	900	1180 (motor-generator set)	900
Motor power, hp	350	500 (motor-generator set)	350
Sealing arrangement	Mechanical shaft seal	Total metal enclosure	Mechanical shaft seal
Material	304 S.S.	304 S.S.	2 1/4 Cr-1 Mo
Type of speed control	Eddy-current coupling	Variable voltage (motor-generator set)	Eddy-current coupling
Manufacturer	Byron-Jackson	General Electric	Byron-Jackson

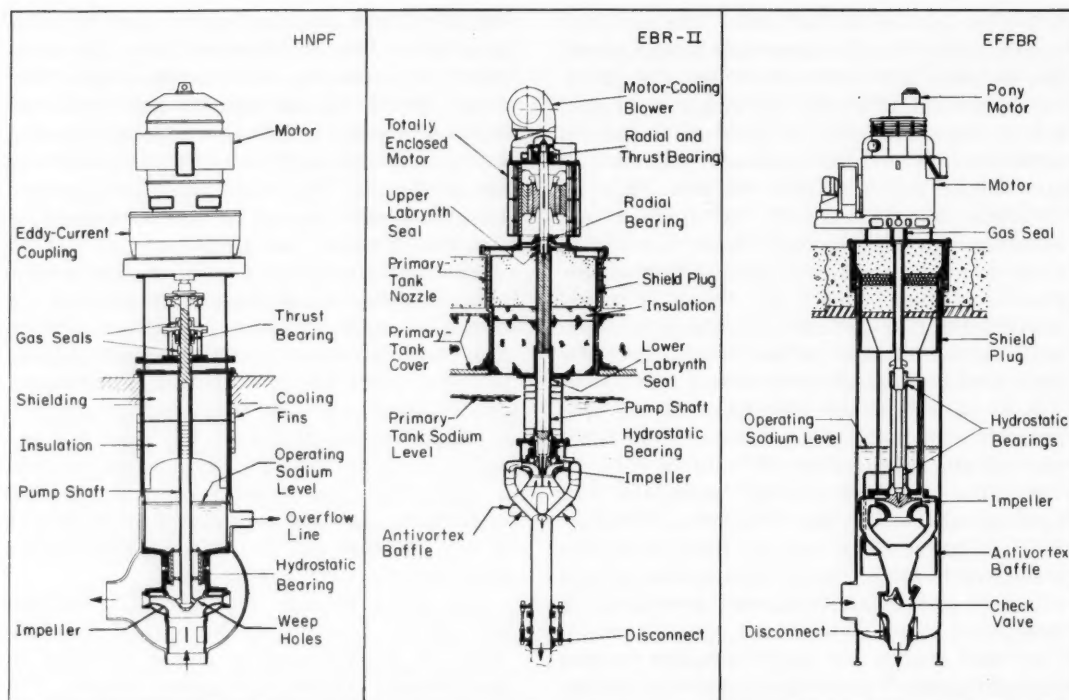


Fig. IX-4 Primary system sodium pumps for HNPf, EBR-II, and EFFBR.¹⁰

reactors are shown in Fig. IX-4. The EFFBR pumps have been operated for more than 7000 hr without difficulties. Some troubles with binding have been experienced with both the HNPf and EBR-II pumps, and, in both cases, insufficient clearances and misalignment (possibly due to uneven temperature distribution in the pump casing in HNPf and to warpage of the three-piece welded shaft in EBR-II) are judged to be at least partially responsible. The difficulties are believed to be incidental rather than basic, and extrapolation of centrifugal pump designs to large sizes is considered feasible.¹⁰

Sodium Graphite Reactors

Status and Operating Experience

The development of the sodium graphite reactor has centered around the SRE, a 20-Mw(t)

experimental plant, and the HNPf, a 75-Mw(e) generating plant operated by the Consumers Public Power District of Nebraska for the Atomic Energy Commission. The SRE began operation in 1957, and most of the nonroutine items of operating experience have been reviewed in past issues of *Power Reactor Technology*. The major problems were those resulting from the leakage of Tetralin into the primary coolant system [*Power Reactor Technology*, 3(2): 60-64 (March 1960); 5(1): 84-87 (December 1961)] and the unexpected power coefficient of reactivity due to the bowing of fuel elements [*Power Reactor Technology*, 6(1): 57-61 (December 1962)]. Both of these problems were incidental rather than fundamental and have been corrected. The first core, of unalloyed uranium metal, was run to a maximum exposure of 1200 Mwd/ton, and the fuel exhibited volume changes of 8 to 9%. Exposures of about 5000 Mwd/ton have been attained with the second core, of metallic thorium-uranium fuel, with little dimensional change. The SRE is now being modified to raise the power level

from 20 to 30 Mw(t) and the sodium outlet temperature from 950 to 1200°F; the new fuel loading will consist of uranium carbide elements designed to operate at central temperatures in the range from 1700 to 2000°F.

The design of the HNPf was reviewed in *Power Reactor Technology*, 5(3): 39-51 (June 1962). The plant first attained the design power level in July 1963. The fuel elements, of uranium-10% molybdenum, have accumulated peak exposures of more than 1200 Mwd/ton. The main problems encountered in operation were the following:

1. Helium was entrained in the main secondary sodium loops from the cover gas in a tank that was provided to accommodate sodium expansion and to provide a low-velocity free surface for degasification. The entrainment occurred because of the formation of a vortex at the exit of the tank. The trouble was corrected by bypassing over 90% of the sodium flow past the tank. This does not negate the effectiveness of the tank for its intended function.

2. Leaks developed in one of the intermediate heat exchangers because of flow-induced vibration of the peripheral tubes. The trouble was corrected by installing support shims in all of the six heat-exchanger units.

3. Some of the Zircaloy-2 control-rod thimbles failed because of hydriding and resulting embrittlement. The hydrogen apparently came from titanium hydride shields placed in the thimbles to counter neutron streaming. The trouble was corrected by replacing the Zircaloy-2 thimbles with stainless-steel thimbles; the resulting reactivity loss was about 0.7%. A circulating-gas and hydrogen-removal system was also added to the control-rod-thimble gas-purge system.

The first of these difficulties, the entrainment of gas in the coolant, appears to be a potential problem to be aware of in the design of sodium systems. Both the DFR³³ and the BR-5 fast reactor⁸ experienced troubles with cover-gas entrainment in the coolant of the primary loop.

Development of Uranium Carbide Fuel

Although uranium carbide fuel¹² may be attractive in other applications, its development has received the greatest impetus from the

sodium graphite reactor program, and forecasts of future developments of the sodium graphite reactor are based on the use of the carbide fuel. The carbide is compatible with sodium and has the advantage, in comparison with uranium metal, of good stability under irradiation. Compared with uranium oxide, the carbide has the advantages of higher density (13.0 g/cm³) and much higher thermal conductivity [about 0.23 watt/(cm)(°C)]. The latter advantages are particularly important in the sodium graphite reactor, where it is better to use rather massive fuel elements in order to minimize neutron losses to cladding and coolant, without sacrificing the high heat fluxes that the cooling properties of sodium make possible.

Ordinarily, uranium monocarbide (UC, 4.80 wt.% carbon) is used, but acceptable performance does not appear to depend upon the attainment of the exact stoichiometric composition. Most experience with the carbide elements has been accumulated with the slightly uranium-rich composition (hypostoichiometric) in arc-cast slugs with sodium-bonded stainless-steel jackets. The results of capsule irradiations are summarized in Table IX-3.

Reactor experience with full-size carbide elements began in June 1963 when two full-length elements were placed in the second core of the SRE. In September 1963, 10 full-size carbide elements began operation in the HNPf reactor. Experience to date with both groups of elements is described as entirely satisfactory.¹² The second full-core loading for the HNPf reactor will consist of 171 carbide elements, essentially the same in design as the 10 elements currently installed. The fuel-element design data for the HNPf carbide elements, and for two projected designs of large sodium graphite reactors (LSGR) are given in Table IX-4. The anticipated performance is believed to be compatible with test experience to date.¹² The design criteria that led to the data given in Table IX-4 are discussed in Ref. 5.

Higher specific powers than those shown in Table IX-4 are thought to be possible—linear power ratings up to 50 kw/ft, corresponding to central temperatures of about 2400°F—but close control of the carbide composition to the stoichiometric value would probably be necessary to yield high-burnup capability in this temperature range.¹²

Table IX-3 RESULTS OF CAPSULE IRRADIATION TESTS¹² OF URANIUM CARBIDE FUEL SAMPLES[†]

Capsule	Cladding	Fuel comp., % C	Annulus width, in.	Bond	Fuel diameter, in.	Actual peak temperature, °C		Average fuel temperature, °C		Burnup (isotopic and/or radiochemical), 10 ³ Mwd/metric ton of U	Increase in Dia., %		Fission-gas release, % ²³⁵ Kr	Cladding carbon content, wt. %		
						Center	Surface	Center	Surface		Length, %	Postirrad. (nom.)		Postirrad. (max.)		
BMI-23-2	304 S.S.	5.3	Wire basket	NaK	0.375 gnd†	1000	580	750	490	5.6	0.9	0.072	Not analyzed			
T ₈		5.1				800	490	570	370	7.8	0.9					
BMI-23-3-B†	304 S.S.	5.0	Wire basket	NaK	0.375 gnd	890	380	590	330	12.8	0.8	0.072	Not analyzed			
BMI-23-5	304 S.S.	4.6	Wire basket	NaK	0.375 gnd	720	535	720	470	8.6	0.8	0.072	Not analyzed			
T		4.55				780	455	670	460	6.4	1.2					
B		4.8	Wire basket	NaK	0.375 gnd	560	780	435		7.0	0.5					
BMI-23-6	304 S.S.	4.55				430	510	315		7.0	0.5					
B		4.4	4.7 to 0.015	Na	0.5 ac**	1340	870	980	675	6.5	2 to 6	2.5 to 3	0.072	0.16		
AI-3-1	304 S.S.	4.4 to 4.7	0.015	Na	0.5 ac	1150	680	1010	555	15.0	1 to 4	0.18	0.10	0.07		
AI-3-4	NI	4.8 to 5.0	0.050††	Na	0.5 ac	1530	920	1150	730	13.0	3.8 to 7.6	21.6	0.072	0.32		
AI-3-5	304 S.S.	4.2 to 4.4	0.015	Na	0.5 ac	970	540			9.1	0.4	0.06	0.072	0.51		
AI-5-1	304 S.S.	4.8 to 5.0	0.005	NaK	0.5 ac	1260	780	940	570	1.4	lost	Gas	0.072	1.26		
NAA-48-1	304 S.S.	4.8 to 5.0	0.005	NaK	0.5 ac	1080	670	480	295	23.9	0.6 to 5.0	1.3	0.072	0.84		
NAA-48-2	304 S.S.	4.8 to 5.0	0.005	NaK	0.5 ac	1110	650	830	500	19.0	3 to 7	7.0	0.072	0.96		
NAA-48-3	304 S.S.	4.8 to 5.0	0.015	NaK	0.5 ac	1290	755	950	575	6.4	3.2 to 4.4	0.6	0.072	1.39		
NAA-48-6	304 S.S.	4.8 to 5.0	0.015	NaK	0.5 ac	1040	600	980	570	1.0	1.0	0.74	0.26			
NAA-81-3	304 S.S.	4.4 to 4.8	0.025	Na	0.5 ac	1150	735	1120	715	39.9	10.0					
UNC 1-1	Nb-1 Zr	4.45	0.020	Na	0.375 ac	1165	800	1020	700	34.8	8.9	31				
		4.70				1180	790	980	660	36.0	5.7	5				
		5.20														
UNC 1-2	Nb-1 Zr	4.45	0.020	Na	0.375 ac	1250	790	1120	720	22.9	11.0	30	Nb-1 Zr cladding			
		4.70								22.9		7				
		5.20								22.9		4				
SU-20	304 S.S.	4.4 to 4.7	0.020	Na	0.5 ac	870	540	Data not reduced		Not examined						
SU-28	304 S.S.	4.4 to 4.7	0.020	Na	0.8 ac	940	690	Data not reduced		Not examined						

* Fuel enrichment was 10% in all capsules.

† Range of heat fluxes calculated at fuel surfaces: 9.15 to 13 × 10⁵ Btu/(hr)(sq ft) (288 to 410 watts/cm²).

‡ gnd, the fuel surface ground after casting.

††, the capsule top.

** ac, the capsule bottom.

*** ac, the as-cast fuel-surface condition.

††† Three 0.050-in.-diameter spacer wires in fuel-cladding annulus.

Table IX-4 DESIGN DATA FOR CARBIDE FUEL ELEMENTS FOR SODIUM GRAPHITE REACTORS¹²

	81-Mw(e) HNPF core I (operating)	200- to 250-Mw(e) LSGR prototype (projected)	1000-Mw(e) LSGR (projected)
Materials			
Fuel (all hypothetically UC)	4.4 to 4.8 wt.% C	4.5 to 4.8 wt.% C	4.6 to 4.8 wt.% C
Cladding	304 S.S.	304 S.S.	304 S.S.
Bond	Sodium	Sodium	Sodium
Enrichment, wt.% ²³⁵ U	3.7 and 4.9	3.8	3.6
Element			
No. of rods	8	18	18
U content, kg/element	146	124	124
Peak specific power, kw/kg of U	18	52	72
Fuel-slug diameter, in.	0.872 (2.22 cm)	0.500 (1.27 cm)	0.500 (1.27 cm)
Active fuel length, ft	13 (3.96 m)	14 (4.26 m)	14 (4.26 m)
Total fuel-rod length, ft	15 (4.57 m)	18 (5.49 m)	18 (5.49 m)
Cladding thickness, in.	0.010 (0.25 mm)	0.010 (0.25 mm)	0.010 (0.25 mm)
Sodium annulus thickness, in.	0.030 (0.76 mm)	0.025 (0.63 mm)	0.025 (0.63 mm)
Design criteria			
Fission-gas release, % of total produced	3	3	3
Maximum cladding strain, %	0.5	0.5	0.5
Peak cladding temperature, °F	1005 (542°C)	1250 (676°C)	
Peak fuel temperature, °F	1600 (870°C)	1750 (955°C)	2000 (1095°C)
Average burnup, Mwd/metric ton of U	15,000	25,000	25,000
Maximum fuel-volume increase, %	17	17	17
Fuel-diameter increase, %	7	5.3 (average)	5.3 (average)
Maximum rod linear power, kw/ft	28 (920 watts/cm)	24 (788 watts/cm)	35 (1150 watts/cm)
Maximum heat flux, Btu/(hr)(sq ft)	207,000 (128 watts/cm ²)	549,000 (173 watts/cm ²)	
Mixed mean temperature, coolant outlet, °F	945 (508°C)	1150 (620°C)	1150 (620°C)

Directions of Future Development

Studies by Atomics International on large sodium graphite reactors are the sources for projections of the direction of development for the reactor type. With regard to performance, the main effort is in the direction of higher specific power (Table IX-4) and higher temperature. Conversion ratios above about 0.5 are not predicted for the reactor type.⁵ Reference 5 states that supercritical steam pressures can be justified economically for plants above 400 Mw(e) capacity, whereas subcritical pressures, from 1800 to 2400 psig, would be used for smaller plants. Table IX-5 shows projected conditions for reactors in the two size ranges. In either case the projection calls for an increase in the primary sodium outlet temperature to 1150°F, from the 945°F used in HNPF.

As mentioned above, the modified SRE is to operate with a sodium outlet temperature of 1200°F.

Perhaps the most annoying problem of the sodium graphite reactor is that of separating the sodium from the graphite moderator. In the SRE and HNPF, the approach is to can the moderator in sizable units in zirconium or steel cans. The advanced design studies for large reactors use a calandria design, with the sodium flowing through the calandria tubes. This approach is expected to permit higher sodium temperatures and higher power density while the parasitic loss of neutrons is being reduced.

Standardization and modular construction are features of the advanced designs which are expected to reduce costs. A common height of

Table IX-5 TYPICAL PROCESS CONDITIONS FOR SMALL AND LARGE FUTURE SODIUM GRAPHITE PLANTS⁵

	200 to 400 Mw(e)	400 to 1000 Mw(e)
Steam conditions, psig/°F/°F (kg/cm ² /°C/°C)	2400/1000/1000 (168.5/538/538)	3500/1000/1000 (246/538/538)
Final feedwater temperature, °F (°C)	473 (245)	512 (267)
Reactor sodium outlet temperature, °F (°C)	1150 (621)	1150 (621)
Intermediate heat-exchanger log mean temperature difference, °F (°C)	75 (24)	90 (31)
Reactor sodium inlet temperature, °F (°C)	750 (400)	750 (400)
Net efficiency, %	41.8	43.6

14 ft is projected for reactors ranging in capacity from 200 to 1000 Mw(e), and a fuel-slug diameter of $\frac{1}{2}$ in. is thought to be near the optimum. A basic loop of 250 Mw(e) capacity is considered to be a practical module for the primary coolant system. The modules for the steam generators are of much lower capacity, 12 to 15 Mw(t), as described in the preceding general discussion of sodium-system components.

Fast-Breeder Reactors

There is general agreement that the fast breeder will be needed in the long term, although opinion varies as to how soon the need will be serious. Most of the countries with major nuclear power programs are pursuing development programs aimed at achieving large, economically competitive, fast-breeder power plants in the next 10 to 15 years. In England and France, where the current nuclear power generation is based on natural uranium, the intention is to start the fast breeders on plutonium from the thermal reactors; and, indeed, some of the incentive for fast-breeder development seems to be the desire to do something with the accumulating plutonium. In the United Kingdom, at least, the plutonium stocks are expected to become large several years before enough fast breeders are in commercial use to absorb them.¹³ In France the accumulation of plutonium is expected to be something over 4 tons by 1975. This amount is considered to correspond approximately to the supply for a first large breeder plant.¹⁴

In the United States and the USSR, where enriched uranium is the fuel for practically all single-purpose power reactors, no great emphasis has been placed on coordinating the development of fast breeders with the growth of the plutonium stockpile. In the United States a considerable analytical effort is devoted to determining the dollar value of plutonium for thermal-reactor fuel, in addition to substantial development efforts aimed at utilizing the plutonium in the thermal-reactor application. In the USSR the economic potential of the fast breeder seems to be regarded more highly than elsewhere, and the forecasts appear to predict its economic superiority even when fissile isotope is not in short supply.

The more favorable Russian estimates may be due, in part, to the different approach used in economic analyses. Instead of trying to evaluate a reactor type by fitting it into an existing complex of mining, transportation, fabrication, processing, and power transmission facilities, the Russians apparently try to evaluate the effort involved in setting up this entire complex for the reactor type in question. This type of approach is also used in evaluating nuclear power vs. coal power. Apparently much of the attractiveness of the fast breeder stems from the expectation that doubling times may be achieved which are short enough to provide for the expansion of nuclear plant capacity without requiring any large expansion of mining operations or enrichment facilities, as indicated by the quotation from Ref. 15:

Most profitable and at the same time providing for the most effective use of nuclear fuel is a development of nuclear power with the priority given to fast 500 to 1000 MWe reactors with a mixed uranium and plutonium fuel cycle and the increase of the proportion of plutonium breeder reactors.

Such a trend would require small scale industries for production and chemical reprocessing of core fuel elements and moderate capacities for production and chemical reprocessing of blanket fuel elements as well as for reenrichment of the core fuel with the plutonium produced in the reactor, enriched fuel being required only to provide the initial load of every new reactor and one or two of its refuelings.

No constant expansion of uranium mining or enrichment industries will be required; nor practically will the power industry be dependent on these, because two or three sets of fuel cores are a sufficient "trigger pulse" for further operation of reactors that would be self sustaining and at the same time provide a certain plutonium excess for new capacities at the rate of eight to ten per cent of annual power increase. This rate can be considerably increased through the use of fissionable materials formerly intended for defense purposes. Such possibilities are opened up in the Statement made by N. S. Krushchev, Chairman, USSR Council of Ministers on the 21st April 1964.

The relatively low estimate of investment in fuel-cycle installations for the fast breeder (includes mining, ore processing, and transportation investment, as well as investment for chemical processing and fuel fabrication) is indicated in Table IX-6, which summarizes some of the results of a Russian study of future nuclear power possibilities.¹⁵ Surprisingly enough, the unit investment in the power station itself is also estimated to be somewhat lower for the fast breeder than for the water-moderated reactor.

Table IX-6 ESTIMATES FOR THE FUTURE* OF REQUIRED UNIT INVESTMENTS AND POWER PRODUCTION COSTS FOR SEVERAL NUCLEAR PLANT TYPES IN THE USSR¹⁵

	Gas-cooled graphite natural- uranium reactors	Heavy-water organic natural- uranium reactors	Water-cooled graphite slightly enriched-uranium reactors with steam superheating in the core	Water-cooled water-moderated slightly enriched- uranium reactors	Fast sodium reactors
Mean unit investment, roubles/kw(e)					
Investments in nuclear power stations	179 (75%)	186 (80%)	114 (66%)	135 (66%)	127 (79%)
Investments in fuel-cycle installations	60 (25%)	46 (20%)	59 (34%)	70 (34%)	34 (21%)
Total investment	239	232	173	205	161
Mean power production cost, kopeks/kw-hr	0.40	0.42	0.43	0.48	0.33

*These estimates are from a study covering the period up to 1980. They are based on plants in the 500- to 1000-Mw(e) range.

Problem of the Large Reactor

Although there has been considerable development in fast breeders since the 1958 Geneva Conference, the significance of an important part of the development has been more in the revealing of basic difficulties than in the solution of them. Specifically, the developments in fast reactor physics and safety analysis, and their application in large-reactor design studies, have shown that there is a fundamental problem in extending the design of sodium-cooled fast breeders from small units to large ones: the problem of preserving an inherently negative power coefficient of reactivity under all credible circumstances.

In small fast reactors it was clear that, once the fear of a substantial positive Doppler coefficient of reactivity due to the fission resonances was allayed, the power coefficient of reactivity could always be made inherently negative, at least so long as the mechanical structure of the core remained intact. The major components of the power coefficient were due to: the linear expansion of fuel elements, which increases neutron leakage by reducing the average density of fuel in the core; the Doppler coefficient, either very small (if there was little fertile isotope in the core) or negative; and the reduction of sodium density by thermal expansion or boiling. The latter was a negative component because its major effect was to increase neutron leakage, and this effect was considerably larger than other, positive effects, such as the effect of neutron absorption by sodium. Experience with EBR-I had, of course, shown that the thermal bowing of fuel

rods could lead to positive contributions [see *Power Reactor Technology*, 5(2): 24-29], but with suitable mechanical design this possibility could be eliminated. The designer could count on a negative power coefficient that, if large and fast acting, could be a substantial help in countering the effects of accidental power excursions or, if small or slow, would at least not compound the effect of such an excursion.

As much larger reactors were investigated, with relatively high ratios of fertile-to-fissile isotope in the core, it became clear that it was very difficult, if not impossible, to preserve the negative character of the sodium-voiding component of the power coefficient. Also, this component could be so large as to override all negative components and increase the reactivity by the order of dollars in some cases of substantial sodium loss from the core. This is partly because the neutron-leakage effect becomes smaller as the core size is increased. However, in the relatively "soft" neutron-energy spectrum that exists in the large dilute core, the "hardening" of the spectrum due to a decrease in moderation by sodium also results in a reactivity increase. The reactor physics aspects of the sodium coefficient and other reactivity coefficients were discussed at length in *Power Reactor Technology*, 7(2).

Once it is clear that a real threat of a net positive coefficient exists, the problems of evaluating it and its significance become very complex. It soon becomes evident, for example, that the overall sodium-void coefficient for the core is not the only consideration. The designer and the safety analyst must consider the local value of the coefficient at the point in the core

where boiling would start first in the event of a power excursion or a cooling interruption. Usually the coefficient is found to be more highly positive in these higher power regions of the core. The process of optimizing the power coefficient through the proper adjustment of design variables is a complex one because all components of the coefficient respond to changes in these variables, and the designer is apt to find, for example, that a design change which reduces the positive nature of the sodium-void coefficient will also reduce the magnitude of the desirable, negative Doppler coefficient. Finally, the evaluation of the coefficients by neutron physics analysis is a very complex problem that still involves large uncertainties.

Although fast reactor theory is, in many respects, the most sophisticated branch of neutron physics, its inadequacies become apparent when it is applied to the evaluation of complex reactivity coefficients. These inadequacies are not due to any deficiencies of fundamental understanding but to the sheer range and detail of data that must be handled (and must be available from experimental measurement). It is still possible that further developments in neutron physics may ease the problem of the power coefficient. For example, the French results summarized in Table IX-7 indicate that calculations which take into account the self-shielding of resonances predict a much more favorable sodium-void coefficient.

It should be made clear that the reactivity-coefficient problem does not appear to be a problem of normal reactor operation, but one that affects the prognosis for the course of a hypothetical accident. The seriousness with which the problem is viewed varies widely from

group to group and from country to country, and reactions vary from a rather casual recognition of the problem to the initiation of searches for alternate fast reactor coolants.

In the United States there does not appear to be any widespread inclination to doubt sodium as the optimum coolant for fast breeders, but the problem of the sodium coefficient of reactivity is taken quite seriously. It appears unlikely that any easy solution of general applicability will be found. However, individual solutions for specific designs will be found through detailed and exhaustive safety analyses. This implies that fast reactor physics, safety experiments, and methods of safety analysis will continue to get much attention in the United States and probably abroad. The current emphasis on these fields is attested by the papers at the Conference, listed as Refs. 16 to 25. The papers are too detailed for review here, but they contain much pertinent and useful information.

Fast Reactor Fuels

Although most of the general considerations involved in the selection and development of fuels for thermal reactors apply also to fast breeders, their relative importance is different. In addition, in fast reactors the effects of the fuel on reactivity coefficients and on reactor behavior in hypothetical accidents must be considered carefully.

Even the largest fast breeders must use fuel of relatively high enrichment. For a given method of fuel processing, high enrichment leads to high processing costs (per unit of fuel comprising fissile-plus-fertile isotope) because

Table IX-7 INFLUENCE OF RESONANCE SELF-SHIELDING ON SODIUM-VOID COEFFICIENT AND BREEDING RATIO¹⁴
(Spherical, 2886-Liter Core with Oxide Fuel; Fuel Volume Is 35% of Core Volume; Fissile Isotope Is ²³⁹Pu; and Fertile Isotope Is ²³⁸U)

²³⁸ U/ ²³⁹ Pu		Sodium-void coefficient,* 10 ⁻⁵ (δk/k)/°C				Internal breeding ratio	Total breeding ratio
		Diffusion	Absorption	Degradation	Total		
Without self-shielding	6.25	-0.736	+0.072	+1.361	+0.697	0.916	1.455
With self-shielding	6.86	-0.775	+0.080	+0.597	-0.098	0.874	1.408

*The calculation breaks the sodium-void coefficient down to the components of neutron diffusion (leakage), absorption by sodium, and the effect of sodium in degrading the neutron-energy spectrum. The calculations show a big reduction in the latter (positive) effect when resonance self-shielding is taken into account. The sodium-void coefficient is expressed in terms of a sodium-temperature coefficient, via the volumetric coefficient of expansion of sodium.

of the criticality problems and the high cost of fuel losses. Moreover, the high enrichment necessitates high specific power if fuel-inventory costs are to be kept within reasonable bounds. This in turn requires a highly subdivided fuel structure that usually involves high fabrication costs. Further increases in processing and fabrication costs (relative to thermal reactors) tend to result from the necessity for recycling plutonium (or ^{233}U) and possibly from the necessity of processing blanket fuel as well as core fuel. The attainment of low fuel cost in the fast reactor therefore appears to require either that inherently cheaper methods of processing and fabrication be found, relative to those used for thermal-reactor fuel, or that the exposure lifetime of the fuel be increased greatly beyond that which is acceptable for thermal reactors. The relative insensitivity of the neutron economy of the fast reactor to the presence of some materials that would absorb neutrons strongly in a thermal reactor assists in both these directions. It permits the use of fuel jackets that are rather thick relative to the fuel element. It also makes the neutron economy rather insensitive to the fission-product burden of the reactor, so that long exposures do not degrade the neutron-physics performance importantly; and it removes the neutron-physics requirement for complete fission-product removal during processing.

It appears to be the result of cost studies that if oxide fuel elements are used in fast breeders, the attainment of acceptable fuel costs requires average fuel exposures of about 100,000 Mwd/ton. Documentation of the conclusion is not easy to find, but it is implied by the frequent use of this exposure in economic studies of fast breeders. Processing and fabrication estimates for oxide elements should be more reliable than estimates for other types, because of the more extensive experience with oxide in the United States. Irradiation tests have indicated that exposure lifetimes of 100,000 Mwd/ton are indeed possible for mixed uranium-plutonium oxide elements of types acceptable for fast breeders.²⁶ Consequently it is tempting to conclude that oxide fuel elements are essentially proved as practical elements for the fast breeder. Certainly it appears that the oxide, although it may not ultimately prove to be best, offers the most straightforward approach to a feasible fast-breeder fuel element. But more extensive irradiation experience is required, and the gap in

experience, between processing and fabrication of uranium oxide elements for thermal reactors and comparable experience with the uranium-plutonium oxide elements for fast breeders, must not be overlooked.

Design characteristics of fuel rods proposed for the Fast Ceramic Reactor (Table IX-8) are

Table IX-8 PROPOSED DESIGN CHARACTERISTICS OF FUEL ELEMENTS FOR THE FAST CERAMIC REACTOR²⁶

Form of element	Rod
Fuel composition	80 wt.% UO_2 -20 wt.% PuO_2 , solid solution
Form of fuel	0.220-in.-diameter pellets, pressed and sintered
Fuel density, % of theoretical density	95
Jacket	AISI 347 S.S. tubing
Jacket outside diameter, in.	0.250
Jacket thickness, in.	0.015
Fuel length, ft	2 to 4, depending on reactor design

considered typical for oxide elements.²⁶ Preliminary estimates of the fabrication cost, for a production rate corresponding to 3000 Mw(e) of generating capacity, yield \$211 per kilogram of uranium and plutonium in the core. This corresponds to a fuel-fabrication contribution to the power cost of 0.29 mill/kw-hr for an average exposure level of 100,000 Mwd/ton.

British work on the irradiation of oxide fuels is summarized in Ref. 27. It calls attention to erratic behavior with respect to fission-gas release, and another British paper²⁸ cites oxide fuel-element designs for fast reactors in which plenum volumes are left in the element, for the accommodation of fission gases, of volume equal approximately to the volume of fuel.

With respect to reactivity coefficients, the Doppler component of the power coefficient is usually strongly negative in oxide fuels containing high proportions of fissile isotopes. This characteristic, which results from the strong dependence of the average fuel temperature on the power level, is of course desirable. It is usually assumed that no important negative component of the power coefficient due to axial expansion of the fuel can be counted on in oxide cores because of the uncertainty as to how an element composed of cracked oxide will expand thermally. The British irradiations have shown much less cracking of PuO_2 - UO_2 samples than of pure UO_2 samples.²⁷ However, it is questionable whether this difference will be

significant from the point of view of the thermal-expansion behavior of the fuel element.

Metallic fuel elements have been used in the fast reactors EBR-I, EBR-II, EFFBR, and DFR. They have many attractive features for the fast-breeder application, but it seems questionable whether satisfactory resistance to radiation damage can be achieved, at least with plutonium-uranium elements.

In the United States the use of metal fuel has been coupled with the concept of pyrometallurgical processing of the fuel, and the EBR-II plant includes facilities for such processing and on-site remote fabrication by casting techniques. Although there are still large uncertainties in the costs of these processes for a large-scale plant—as indeed there are also uncertainties in the costs of processing and fabrication of oxide fuels for the fast breeder—the current U. S. paper on fast-breeder fuel economics²⁹ does not forecast substantial cost advantages for them. Consequently it appears that the exposure goals for the metal fuels must lie at least as high as 40,000 to 60,000 Mwd/ton, or, roughly, 4 to 6 at.% burnup.

The uranium-fissium* alloys show at least some encouraging performance under irradiation: at 600°C, for example, uranium-5 wt.% fissium alloy shows a swelling rate of about 8% volume increase per atomic percent burnup.³⁰ However, uranium-plutonium-iron alloys swell badly even at relatively low temperatures. The alloy uranium-20 wt.% plutonium-10 wt.% fissium, for example, increases about 20% in volume per atomic percent burnup at an irradiation temperature of 400°C (Ref. 30). It has been found possible, however, to restrain the swelling by the use of sufficiently strong jackets on the metallic fuel. Once the fuel swells to the point that it begins to press against the jacket, the internal pressure generated within the fuel by fission gases must be resisted, essentially by the strength of the jacket alone. The exposure lifetime then depends on the tensile strength of the jacket material, the diameter and thickness of the jacket, and the fractional void volume built into the element as an allowance for radial swelling. In

its simplest form such an element is conceived as a rod of the fuel material inside a tube of the cladding material, with a sodium-filled annular space (initially) between fuel and jacket. Figure IX-5 shows the calculated effects of cladding-wall thickness and radial clearance (given in terms of the inner diameter of the jacket for a constant fuel diameter) on the achievable burnup for uranium-plutonium-iron elements.

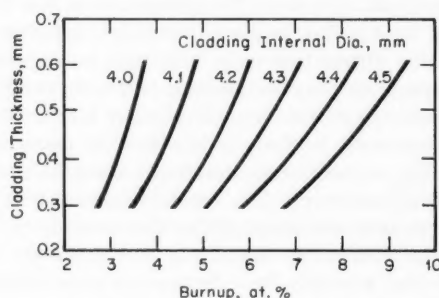


Fig. IX-5 Calculated effect of cladding thickness and internal diameter on the achievable burnup in 3.65-mm diameter uranium-plutonium-fissium alloy rods under EBR-II temperature conditions. A maximum allowable cladding stress of 30 kg/mm² was assumed.³⁰

Thorium-rich thorium-uranium alloys show relatively good resistance to radiation-induced swelling, even at temperatures up to 750°C. Although irradiation results are not available at high temperatures for thorium-plutonium alloys, the few results that have been obtained at 450°C are promising.³⁰

Although stainless steel was used as the fuel-jacket material in EBR-I, it has the undesirable property of forming a eutectic alloy with uranium, of relatively low melting point. As jacket materials for uranium metal, the nickel-base alloys have the same drawback. Of the refractory metals that do not present this problem, only vanadium appears completely satisfactory as an alloy base with respect to corrosion resistance and nuclear properties.²⁶ Currently, an alloy of vanadium-20% titanium appears very promising. Its fabrication into high-quality thin-walled tubing is difficult, however.

In the DFR, metallic fuel elements have been used in a number of design variations, but basically the elements consist of annuli of

*The term "fissium" is used to designate the mixture of fission products, comprising molybdenum, ruthenium, technetium, palladium, zirconium, rhodium, and niobium, left in the exposed fuel after pyrometallurgical processing.

uranium alloy with strong inner and outer jackets of niobium, originally designed to restrain swelling. The outer diameter of the annular element is approximately 0.75 in. The thicknesses of the fuel annuli have been varied during the development of the elements. The elements currently in the reactor use uranium-20 at.% molybdenum as the fuel alloy, with annuli 0.10 in. thick for the central region of the reactor core and 0.13 in. thick for the peripheral core regions. The elements are vented to allow those fission gases which may be released by the fuel to escape to the coolant. Irradiation tests have shown that the temperature of this fuel should be limited to 650°C if rapid swelling is to be avoided. Furthermore, it has been found that there is a lower temperature range, from 480 to 580°C, where rapid swelling of uranium-molybdenum alloys can occur during irradiation because of a transformation of the metastable gamma phase to alpha-plus-delta phases. If the fission density is high enough, however^{30,31} [above 8×10^{13} fissions/(cm³)(sec)], the formation is suppressed and irradiation stability does not suffer. One of the design goals therefore was to hold the fuel temperature below 480°C so long as the fission density is below 8×10^{13} fissions/(cm³)(sec). The DFR experience has shown that these elements will operate satisfactorily up to a burnup of 1.2%, the maximum yet experienced, and probably considerably higher. These results are for a peak specific power of 205 watts/g. The niobium jacket material has shown some incompatibility with the sodium coolant. It has absorbed hydrogen from the coolant with resulting severe embrittlement when the jacket temperature is reduced to a low value. There is, however, a brittle-ductile transformation at 150 to 200°C, and therefore the properties are not impaired at operating conditions.³¹ There is some corrosion of the niobium which is attributed to oxygen. The oxygen level in the DFR coolant has been about 10 ppm. A jacket temperature limit of 480°C is currently imposed on the elements to avoid this corrosion.

If metallic fuel bodies could be counted on to retain their structural integrity, they would offer the favorable property of a substantial negative component of the power coefficient of reactivity due to axial thermal expansion. Although this is probably the case early in fuel lifetime, it is questionable whether the property

is retained after long exposure. In particular, if the jacket is relied upon to restrain fuel swelling, it seems improbable that the fuel would expand axially in proportion to its temperature and its normal temperature coefficient of expansion. Rather, it might be expected that the axial expansion would be governed by the jacket temperature, which is relatively insensitive to changes in power level. Compared to the ceramic fuels, the metallic fuels are unlikely to provide a large reactivity sink due to the Doppler effect before the fuel temperature rises to the point of fuel-element failure. This is particularly the case if the jacket material can form a low-melting eutectic alloy with the fuel.

The carbide fuels have been discussed briefly in the preceding paragraphs, primarily in connection with sodium-cooled thermal reactors. When these fuels are considered for fast reactors, interest is concentrated on the mixed uranium-plutonium carbides and on performance at very high levels of burnup. To date the irradiation results on uranium-plutonium carbides appear to be relatively few and to be limited to exposure levels of 30,000 Mwd/ton or lower.^{12,28} However, the results have been encouraging and suggest that, at any given temperature, the rate of fission-gas release from carbide fuel is similar to or less than that from oxide.²⁸ If the temperature limits of the carbide are indeed comparable to those of the oxide, the carbide should allow the use of much more massive fuel-element designs, because of the high thermal conductivity of the carbide.

It appears that, for the fast reactor application, there may be considerably more emphasis on attaining the nominally stoichiometric composition of the monocarbides. It is pointed out in Ref. 28 that a more appropriate description of the desired composition is that of a single-phase alloy, since there may be small quantities of oxygen and nitrogen present as well as carbon. The desired composition is one in which the sum of the concentrations of oxygen, nitrogen, and carbon is close to 50 at.%. The hypostoichiometric (uranium-rich) compositions are to be avoided because it is believed that the accumulation of fission gas in any free metal that may be present will determine the swelling rate of the fuel body. The hyperstoichiometric compositions show poor compatibility with the common jacket materials

because of the transfer of carbon from the fuel body to the jacket.

The good conductivity of the carbide makes it worthwhile to consider elimination of the thermal resistance of the fuel-jacket gap through the use of some kind of thermal bonding. Sodium bonding may be used, but it accentuates the transfer of carbon from hyperstoichiometric carbides to the jacket. In the United Kingdom some development is being carried out on the metallurgical bonding of carbide fuel to the jacket.²⁸ There has been some success with copper- and nickel-based brazing alloys as the bonding material between uranium carbide and steel jackets, but there is some problem of cracking of the carbide under thermal cycling in the bonded elements.

Another approach to the fast reactor fuel problem is through the use of cermet. These are dispersions of fuel materials (fissile or fissile plus fertile) in ceramic form in a metallic matrix. The ceramic is usually the oxide of the fuel material. The basic concept is to take advantage of the irradiation stability of the ceramic while approximating the thermal conductivity and mechanical properties of a metal. It is desirable that the size of the ceramic particles in the dispersion be sufficiently large that only a small fraction of the fission gas can escape from the particles by fission recoil.

One of the main disadvantages of the cermets is that the ceramic component cannot usually be made to constitute the major fraction of the total material. Therefore, if both the fissile and fertile components of the fuel mixture are incorporated in the ceramic form, the amount of fertile material that can be incorporated is severely limited. This means that the internal breeding ratio of the reactor will be low, that the reactor will tend to lose reactivity rapidly with burnup, and that this reactivity loss must be taken care of either by strong control elements or by frequent partial refueling. The possibility of avoiding this difficulty by using the fertile element as the metal matrix and incorporating only the fissile element in the ceramic form is referred to in Ref. 26. This appears to be an attractive possibility from the point of view of nuclear performance, but restriction of the choice of the metal matrix to uranium, and possibly thorium, is a severe metallurgical limitation. Apparently there has been no extensive work on this concept.

The British have under way a program on cermet fuels³² which is directed primarily toward the development of a fast reactor element and which has concentrated on the use of oxides in a steel matrix. The fissile and fertile materials are mixed in the oxide form, and the problem of achieving an acceptably high fertile content is attacked simply by attempting to use the highest possible oxide content in the cermet. The practical limit of oxide content appears to be about 50 vol.%.

The British experience has shown that, although the cermets can be formed by pressing and sintering the metal-oxide mixture, the densities attainable by this process alone are no higher than about 85% theoretical and that mechanical working is necessary to attain higher densities. Both hot rolling and hot swaging have been used for densification. Pin type elements are produced by mixing the oxide and steel particles, cold pressing into pellets and sintering at 1100 to 1300°C, sealing the pellets into a steel tube, and swaging at a temperature between 1000 and 1300°C (Ref. 32). Conceptually the objective in fabrication of the cermet is to produce an assemblage of oxide granules, each surrounded by a small, steel pressure vessel, and the whole bonded together as a structure. It is fairly obvious that this objective will be promoted if the oxide granules are spherical in shape, and the process does include the use of such granules. The spherical particles are produced by first pressing the oxide into green compacts about 1 cm in diameter with aluminum stearate binder. The green pellets are broken down to granules and sized, and the granules are spheroidized by abrasion in a gyrating dish coated with silicon carbide powder. The spherical particles are sintered at a peak temperature of 1600°C in an atmosphere of cracked ammonia. A further improvement in the cermet has been made by coating the sintered oxide spheroids with an organic binder that makes them sufficiently sticky to cause a layer of stainless-steel powder to adhere to each particle before the cermet is pressed. The material preferred for the metal matrix is type 316L (low carbon) stainless steel.

Plate type dispersion elements have been made by processes analogous to those used for the pins but with hot rolling substituted for the swaging process. Cermets have been made with both uranium oxide and plutonium-uranium oxide

mixtures and have been irradiation tested. The results with the plutonium fuels have not been as good as those with uranium, but the irradiation program has necessarily lagged behind the fabrication development, and the latest improvements have not yet been thoroughly tested. Results³² to date suggest that the fabrication technology is capable of producing cermet plates and pins with 30 to 50 vol.% ceramic that will achieve a burnup of at least 10% heavy atoms without failure when the surface temperature is about 625°C.

Current Status and Future Directions

The fast reactors currently in operation or definitely planned are listed in Table IX-9. Since the Second Geneva Conference, reactors that have contributed new operating experience at substantial power levels are the British DFR⁷ and the Russian BR-5.⁸ In the United States the EBR-I continued in operation, and its latter-day operation was directed primarily to the rather specialized, but necessary, investigation of dynamic characteristics. These EBR-I results have been covered rather thoroughly in *Power Reactor Technology* in the past.

The escalation of power of the DFR to the design value of 60 Mw(t) was a long, troublesome process, which covered the time period from early 1960 to July 1963. It yielded, however, much useful information on sodium technology and metallic fuels.³³ The early problems had mainly to do with the entrainment of gas in the coolant and with the removal of gross impurities from the primary coolant system. The latter problem was finally solved by the use of hot-dumping operations in addition to the more usual cold-trapping processes. A "hot dump" consisted of heating the primary coolant to 250 to 300°C, transferring it to the dump tanks, cooling it to 80°C, and then returning it to the primary system through a filter. Twelve of these dump sequences were used after the fuel had been removed from the reactor.

A more fundamental obstacle to power escalation was the fuel performance, and full power was attained only after the original fuel elements had been replaced by redesigned elements. The original concept for the annular elements had been to accept a fuel alloy of relatively poor swelling resistance, to restrain the radial swelling by the walls of the element,

and to accommodate the volumetric swelling by allowing the fuel material to extrude axially within the confines of the element walls. Early experience showed, however, that the axial extrusion was opposed by frictional forces great enough to exceed the creep strength of the jacket. Redesign of the element involved the substitution of a fuel alloy of higher swelling resistance (uranium-20 at.% molybdenum), reduction of the thickness of the fuel annulus to limit the maximum alloy temperature to 650°C, and the provision of a larger fuel-jacket clearance to allow for more radial swelling. The limit on the alloy temperature and a 480°C limit imposed on the jacket temperature to avoid corrosion have prevented attainment of the design value of coolant outlet temperature. It is suspected that the corrosion of the niobium jacket is accelerated by some impurity in the coolant, possibly carbon, which may be eliminated with time.

Although a number of problems had to be overcome to attain full power, all of them were recognized long before they resulted in operating difficulties, and operation has been smooth and without untoward incident within the recognized limits of the plant. The reactor is continuing operation and is being used extensively for the irradiation of improved fuel elements.

The BR-5 reactor, although much smaller than the DFR, has given valuable results on the behavior of plutonium oxide fuel elements, as well as information (discussed earlier) on the behavior of fission products in sodium systems.

The steel-jacketed, oxide fuel elements of BR-5 are described in Table IX-10. The reactor core operates at an average power density of about 300 kw/liter. The initial fuel loading of plutonium oxide elements operated to an average burnup of 2.4% of the plutonium before any evidence of fission-product leakage was detected. Thereafter the reactor operation was continued, with increasing evidence of fission products in the primary coolant and finally with plutonium alpha activity in the coolant, but without operating difficulty, to a maximum burnup of 5% (see Table IX-1). During this period the primary coolant outlet temperature had been raised from the initial 400 to 450°C level to 500°C. Thereafter the fuel assemblies containing gross failures were replaced by new assemblies of UO₂, and operation was continued at lower temperatures (outlet temperatures of

Table IX-9 FAST REACTORS IN OPERATION, OR UNDER CONSTRUCTION, OR DEFINITELY PLANNED*

Reactor	Country	Power, Mw(t)/Mw(e)	Status	Fuel	Remarks	Reference
Rapsodie	France (with Euratom)	20/none	Being built; full power expected before end of 1967	PuO ₂ -UO ₂ pressed and sintered pellets, 5.7 mm in diameter, in 316L S.S. cans with 0.45-mm wall	The French program is directed toward 1000-Mw(e) fast breeders. It is expected that an intermediate size plant of 80 Mw(e) will be needed between Rapsodie and the large plants. ¹⁴ A large critical experiment (Masurca) and a source-calibration pile (Harmonie) are also under construction at Cadarache	34 34
DFR	U. K.	60/14	Operating at full power since July 1963	U-20 at % Mo alloy in Nb jacket	The reactor is being used primarily to investigate the behavior of various Pu-based fuels and cladding materials. The construction of a prototype large enough to demonstrate the core and fuel characteristics for a 1000-Mw(e) commercial plant is planned by the U. K.; this should be on power soon after 1970. ¹³ Two large critical assemblies are used in the U. K. program, Vera and Zebra. ¹⁵	7 33
BR-5	USSR	5/none	Operating at full power since July 1959	PuO ₂ in S.S. jackets (See Table IX-10)	The previously announced BN-50 reactor has been eliminated from the USSR program, but plans have been announced to construct the BOR reactor, a 50- to 60-Mw(t) experimental fast reactor. Critical experiments for large dilute reactors are carried out at the "BFS"-stand. ²¹	8
BN-350	USSR	1000/350	Definitely planned but schedule not announced	PuO ₂ in S.S. jackets (See Table IX-11)	Although the U. S. has a large program directed to the development of large fast breeders, no commitment has yet been made for the construction of a prototype. The program includes critical experiments on two facilities (ZPR-III and ZPR-VI) ²⁰ and fuel-meltdown studies with the TREAT reactor. ²⁵ Two experimental reactors are under development: FARET, primarily for the test of fuel systems at high temperature, and SEFOR, a 20-Mw(t) developmental reactor using PuO ₂ -UO ₂ fuel ²⁶ (the latter project has participation also by Gesellschaft für Kernforschung mbH and Euratom)	11
EBR-II	U. S.	65/20	Critical November 1963. Power escalation under way	0.144-in. U-Fs alloy rods in type 304 S.S. jackets, Na bonded		9
EFFBR	U. S.	300/100	Critical August 1963. Power escalation under way	0.158-in. U-10% Mo rods clad with metal-lurgically bonded Zr cladding		

*This table lists only the reactors that were discussed at the Conference. It does not include some of the older reactors, such as EBR-I.

Table IX-10 CHARACTERISTICS⁸⁶ OF FUEL ELEMENTS AND ASSEMBLIES IN THE FIRST LOADING OF BR-5

Fuel element	
Type	Rod
Fuel	Sintered plutonium oxide pellets
Jacket material	Stainless steel, type IX18H9T
Jacket outside diameter/ thickness, mm	5/0.4
Fueled length, mm	280
Thermal expansion space (at top of fueled section), mm high	2 to 3
Filling gas	Helium at atmospheric pressure
Fuel assembly	
Type	Hexahedral stainless-steel tube
Dimensions, mm	26, between parallel sides; 0.5, wall thickness
No. fuel elements per assembly	19
Method of supporting elements in assembly	Top and bottom grids

300 to 430°C) and various power levels to study the effects of fission-product contamination of the primary circuit. By Mar. 1, 1964, the maximum burnup in the original plutonium oxide elements that were still in the core was 5.8%.

At the time of fuel-assembly replacement, all assemblies were tested by pumping gas from them and checking it for radioactivity. Some showed activities thousands of times greater than the average. These were discarded (18 out of a total of 81) and replaced by the new UO₂ assemblies. The following quotation from Ref. 8 describes the condition of elements from the "discarded" and "undiscarded" assemblies:

...Two of the discarded subassemblies with a 4.9% burnup (one was processed with steam) and one subassembly with a 5% burnup out of the undiscarded were examined in the hot laboratory. Longitudinal cracks 8-10 cm long and up to 1.0 mm wide were detected on the surface of some elements of these subassemblies. The cracks, as a rule, continued in the plutonium dioxide. Mostly radial cracks were observed on the surface of cross sections of the disintegrated elements. No central cavities were found in any of the disintegrated elements. In the third subassembly examined all the elements turned out to be tight, the element canning was in good condition: the element surface was light-colored and bright and there were no visible defects.

In late 1964 and early 1965 it is planned to load the BR-5 with uranium monocarbide fuel.

It is stated in Ref. 8 that the operation of the BR-5 reactor has already confirmed the practical feasibility of the reactor type for industrial installations. The announced decision¹¹ to bypass the previously planned 50-Mw(e) commercial plant BN-50 in favor of pushing on directly to the large 350-Mw(e) plant (BN-350) appears to be in harmony with this point of view. However, Ref. 11 also announces a project to construct a "fast pilot reactor," designated BOR, for general development work leading to large fast breeders. This reactor, of thermal capacity 40 to 60 Mw, will test advanced concepts and operating conditions: power densities up to 1500 kw/liter, sodium outlet temperatures to 630 to 650°C, and fuel burnup to 10% with thin-walled fuel elements that allow fission-gas leakage to the coolant. This suggests that the elimination of the "commercial" BN-50 may have been the result of a decision that the effort could be spent more effectively on a reactor of more experimental nature.

With the beginning of operation of EBR-II and the EFFBR, fast reactor progress in the United States should accelerate. Neither of these plants can be considered the prototype of an economic commercial fast breeder. Although the EFFBR will operate as a utility station, its fuel is far from satisfactory even on a non-recycle basis. The shortcomings of the fuel have restricted the operating power level, in a manner reminiscent of the early DFR experience. The situation is summarized in the following quotation from Ref. 9:

The present core was de-rated from 300 to 200 MW(t) because of an increase in the estimated pressure drop across the core resulting from a change in the fuel element support arrangement (see Fig. 6, P/2427). Experimental data obtained subsequent to this decision show that the pressure drop will be only 40 psi at 200 MW(t) whereas the subassembly and internals are capable of operating at 65 psi. A higher power level, therefore, is possible but cannot be justified economically because of the burnup limitation on the U-10 w/o Mo fuel alloy. As described in another paper, this alloy has not proven to have the irradiation damage resistance which early results suggested. It has been found that the allowable burnup decreases with increasing fuel temperature. Therefore, after demonstration of 200 MW(t) operation, it is planned to operate at an inlet sodium temperature of 450°F and at a power level of 110 MW(t) which is estimated to double the allowable burnup. Although these conditions are uneconomical for electric power generation, it will extend the fuel life of the two core loadings now on hand and permit use of

the reactor as a test bed for irradiation of experimental fast reactor fuels. Plans are proceeding to utilize the reactor initially for this purpose.

Although the British hope to have a prototype of a commercial fast reactor on power soon after 1970, the date is contingent on the development of a satisfactory fuel and decisions as to the main reactor design features.¹³ Apparently only the USSR is committed far enough to a prototype (BN-350) at this time to announce its general features. These are summarized in Table IX-11. However, the construction schedule for the plant appears to be somewhat

breeders might be developed utilizing different basic fuel types. But for the moment it would be encouraging to be able to verify the completely satisfactory performance—in relation to breeding ratio, reactivity coefficients, and reactivity lifetime, as well as to the more obvious criteria such as fuel lifetime and fabrication cost—of any one fuel. Because of the growing interest in oxide fuels, the coming operation of Rapsodie in France^{34,37} and of the Southwest Experimental Fast Oxide Reactor (SEFOR) in the United States³⁵ will be awaited with interest.

Table IX-11 CHARACTERISTICS OF THE BN-350 REACTOR¹¹

Thermal power, Mw	1000
Core volume, approx., liters	2000
Average power density, kw/liter	500
Sodium volume in reactor vessel, m ³	165
Primary sodium temperature inlet/outlet, approximate, °C	300/500
Maximum sodium velocity in core, m/sec	10
No. of core fuel assemblies	211
No. of blanket assemblies	500
Axial and radial blanket thickness, cm	60
Core diameter/height, m	1.5/1.06
Sodium volume fraction in core, %	39
Fuel	Plutonium and uranium dioxide mixture (19% Pu)*
Critical mass, kg of Pu	780 (or 950 kg ²³⁵ U)
Fuel element	Rod type; oxide pellets in stainless-steel tube, as in BR-5†
Diameter/thickness of jacket tube, mm	5/0.4
Fuel assemblies	96 mm hexagons with wall thickness of 2 mm
No. of fuel elements per assembly	217
Method of spacing elements in assembly	Top and bottom grids plus ribs on element jackets
Breeding ratio, total/internal	1.5/0.62
Rate of reactivity loss, % k_{eff} /month	0.62 (one month gives about 0.5% burnup)
Control rods	Assemblies similar to fuel assemblies with enriched B ₄ C in place of fuel

*The first loading may employ enriched UO₂ (23% ²³⁵U).

†It is stated that the average oxide density will be lower than in BR-5 to accommodate fission gases.

flexible, and it is not clear whether the current concept has lasting significance.

The key question in predicting the course of fast-breeder development is the selection of fuel type. Although the early experience has been predominantly with metal fuel, that experience has not been highly encouraging even for uranium metal, and the prospects for plutonium appear considerably less favorable. On the basis of current knowledge, the oxides appear to be the surest bet as potentially acceptable fast reactor fuels. The carbides, however, have inherent advantages in thermal conductivity and density, and it is primarily the lack of experience with them that puts them in a less favored position. It is, of course, conceivable that a number of acceptable fast

In whatever direction the fuel development goes, considerations of reactivity coefficients are expected to have an important influence on the design of large fast breeders. In particular, there will be efforts to attain a negative value for the sodium-void coefficient of reactivity or minimize its positive value and to provide as large a sink of reactivity in Doppler effect as possible. The effects of these considerations on reactor design are discussed in Ref. 24 and were reviewed at length in *Power Reactor Technology*, 7(2): 107-144.

Partly because of the difficulty with the sodium coefficient of reactivity and partly to circumvent problems and difficulties in sodium technology, some groups have investigated other coolants for fast reactors. The Karlsruhe proj-

ect is investigating cooling by helium and dry steam as well as by sodium,³⁸ and papers on gas cooling³⁹ and steam cooling⁴⁰ were also presented at the Conference by Swiss and Belgian groups, respectively. The latter is a direct-cycle system that utilizes dry steam as the reactor coolant. Part of the superheated steam from the reactor exit goes to the turbine, and another appropriate fraction is directed to a Loeffler boiler where it is quenched in the boiler water and produces dry saturated steam to feed the reactor inlet.

One of the problems with less efficient coolants is to design for the characteristic high specific power of the fast reactor. This may be not so much a problem of normal cooling as of emergency shutdown cooling, particularly upon loss of coolant pressure. In the gas-cooled concept³⁹ it is proposed to use water injection as a last-ditch emergency measure. Criticality can be avoided under these conditions by incorporating resonance absorbers in the reactor core. This would be relatively ineffective in the normal "fast" neutron spectrum but would absorb strongly if the spectrum were degraded by water injection. The problem sounds like a difficult one.

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Section

X

Power Reactor Technology

Maritime Reactors

Of the two nuclear-powered ships that are in operation, only the icebreaker *Lenin* was formally discussed at the Conference.^{1,2} Other papers concerned the nuclear plant for the German nuclear-powered vessel *Otto Hahn*³ and design studies of the VULCAIN⁴⁻⁶ and NERO⁷ reactors for marine propulsion.

The icebreaker is powered by three pressurized-water reactors, each capable of producing 90 Mw(t), although the vessel can be operated at design conditions with only two of the three reactors operating. The core of each reactor consists of 219 fuel assemblies arranged as shown in Fig. X-1. Each assembly is composed of 36 fuel pins fabricated of UO₂ pellets clad with zirconium alloy tubing having an outside diameter of 6.1 mm and a thickness of 0.75 mm. The fuel pins are contained in a flow tube that is fabricated of zirconium alloy, bearing ¹⁰B as a burnable poison. The amount of poison is varied from tube to tube to reduce the radial power-peaking factor. The reactor is equipped with safety and control rods as well as a so-called "compensation system." Although details of the construction of these control components are not given in Ref. 1, a paper from the Second Geneva Conference⁸ may be consulted for additional information.

Each reactor of the *Lenin* is equipped with two coolant loops, each containing a steam generator, two main circulating pumps, an emergency circulating pump, two electrically heated pressurizers, and an ion-exchange filter. Figure X-2, which is a photograph of part of the Russian exhibit, illustrates the reactor compartment of the ship. Although call-outs are not provided, the reader can see the positions of the three reactors in the center of Fig. X-2 as evidenced by their control- and safety-rod "thimbles" protruding above the upper shield structure. The six main coolant pumps and a cutaway showing one of the steam

generators can be seen at the left. The loops are designated "bow" or "stern" loops in the reference, the reactor compartment in Fig. X-2 being arranged with the center line of the pumps in a fore-aft orientation. Table X-1 lists additional operating data for the reactors.

In general, the operation of the *Lenin* has been "quite successful." Much of the success perhaps results from the fact that the icebreaker application approximates an "ideal" application of nuclear energy. The vessel became operational in December 1959 and operated on its first fuel for over 11,000 hr, producing about 450,000 Mw(t)-hr. The average burnup reached about 12,000 Mwd per ton of uranium, with peak burnup reaching about 30,000 Mwd per ton of uranium. Refueling was done in the spring of 1963, and in May 1964 the vessel was operating with the second fuel charge. In 1961 the self-regulation of the reactor was demonstrated using the Doppler effect to com-

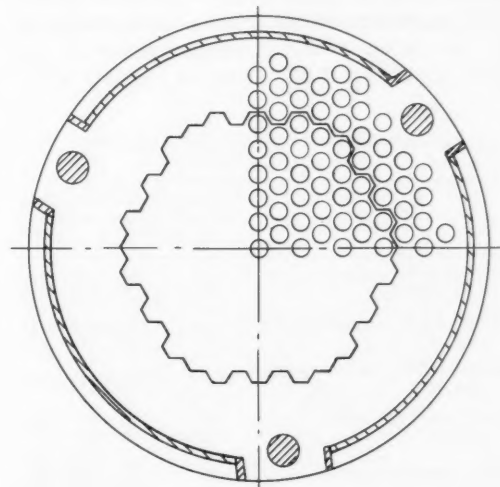


Fig. X-1 *Lenin* reactor core cross section.²

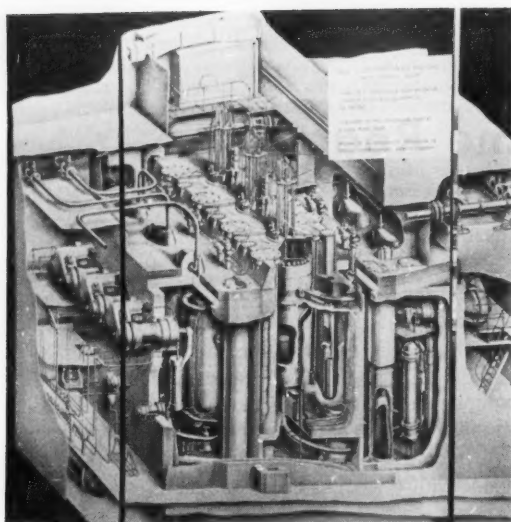


Fig. X-2 Photograph of part of the Russian Geneva exhibit illustrating the reactor compartment of the Lenin.

pensate for load changes. The primary components of the plant operated about 15,000 hr (including the year 1963) at working pressure (180 kg/cm²). The main circulating pumps have operated up to about 8500 hr without inspection, although some pumps apparently have failed owing to shorting of the stator windings.

The primary circuit components that are permanently or periodically filled with potentially contaminated water are accommodated in the so-called "central compartment" of the plant. In Fig. X-2 this probably consists of the

compartments containing the reactors and steam generators, and these compartments are hermetically sealed from the rest of the ship and ventilated separately. Some leakage has been experienced from the primary system. Reference 2 notes that this has come from the packing of various valves in the primary circuit. Contamination from fission gases and long-lived fission products is reported to be low.

Reference 9, although not specifically directed toward the *Lenin* reactors, pertains to a control system that might be uniquely applied to the reactor type. The essence of the system is a reactor in which the coolant is separated from the moderator; the *Lenin* reactors are of this type with their flow-tube design. The principles of the control system are contained in the following quotation:⁹

Let us consider a reactor core formed by a regular lattice of cylindrical fuel elements placed in a liquid moderator, say in H₂O. Let the top ends of all elements be fastened in some plane thus that any of them can turn independently around the fastening point. Let, further, these elements pass freely through the spacer plate, the holes in it forming a grid similar to that of fastening points. The lattice and the form of the cylindrical reactor do not depend on the distance between the plane of joints (fastening points) and that of the spacer plate. Let the plate be rotated in the horizontal plane around the reactor central axis. Then the core will have the form of a one-sheet hyperboloid of revolution and the fuel element lattice will be deformed thus that the surfaces of the constant spacing will turn out to be, with a greater accuracy, ellipsoids of revolution having one focus with the hyperboloid that forms the side boundary.

The reactivity dependence on the core spacing in a cylindrical reactor is not monotone, it has optimum. When the spacing increases reactivity rises first, and then falls. Therefore to control a reactor by means of the core deformation it is possible to use either the "left" branch of the curve corresponding to the core spacing less than optimum or the "right" one corresponding to the spacing greater than optimum.

The reactivity of a reactor controlled in the above manner was studied both analytically and experimentally, and the reference may be consulted for details.

The power plant for the German research ship *Otto Hahn* is described in Ref. 3. It will be a pressurized-water reactor (PWR) having the characteristics shown in Table X-2. The ship was launched in June 1964, and the time schedule calls for the reactor to be installed in 1966 with power operation to commence in 1967. The normal power of the propulsion plant is 10,000 shp.

Table X-1 OPERATIONAL DATA FOR THE LENIN REACTORS²

	Designed values for 65 Mw	Operation data for reactors 1, 2, and 3*		
Water flow in primary circuit loops, tons/hr	415	435 430	458 467	435 453
Reactor outlet temperature for loops, °C	317	311 311	312 313	311 313
Reactor inlet temperature for loops, °C	261	260	261	260
Steam output for loops, tons/hr	43.3 43.3	42 42	47 42	43 46
Steam pressure, kg/cm ²	29	32 31	31.5 30.5	31 31
Steam temperature, °C	307	310 309	308 308	308 308

*Fraction numerators refer to bow loop and the denominators to stern loop.

Table X-2 MAIN DATA OF REACTOR INSTALLATION FOR THE SHIP OTTO HAHN³

Core	
Diameter of active zone, mm	1150
Height of active zone, mm	1120
No. of fuel elements	16
No. of control rods	12
Fuel elements (side length), mm	270
No. of fuel rods installed	3128
Fuel-rod diameter, mm	10.9
Fuel-rod cladding thickness, mm	0.33
Total surface, m ²	120
Core entrance temperature, °C	266
Core exit temperature, °C	278
Maximum thermal output, Mw	38
Average reactivity lifetime, Mwd/ton	7200
Maximum reactivity lifetime, Mwd/ton	14,000
UO ₂ fuel loading, tons	2.95
Enrichment in four zones, %	~2.5, 2.9, 3.5, and 4.3
Average enrichment, %	~3.6
Surface heat release, kcal/(m ²) (hr)	272,000
Average fuel temperature over lifetime, °C	460-680
Hot-channel load at central fuel melting, %	172
Absorbing material in control rods	Boron carbide in steel
Burnable poison	ZrB ₂ -ZrO ₂ pellets
Excess reactivity (cold, unpoisoned), %	15
Excess reactivity (hot, unpoisoned), %	7
Excess reactivity (hot, poisoned), %	4.5
Shutdown reactivity of control rods, %	25
Pressure vessel	
Operating pressure, ata	63
Inside height, mm	8580
Inside diameter, mm	2360
Wall thickness and plating, mm	50 + 5
Coolant circulation, kg/hr	2.37 × 10 ⁶
Steam generator	
Operating pressure, ata	31
Steam temperature, °C	273
Superheat, °C	36
Tube diameter, mm	19.4
Tube wall thickness, mm	1.2
Material	Inconel
Containment vessel	
Height, m	13.1
Diameter, m	9.5
Wall thickness, mm	30
Design pressure, ata	14.5
Weight including reactor parts, tons	930

The German maritime reactor³ incorporates some of the advanced techniques described in the review of the Consolidated Nuclear Steam Generator and the Unified Modular plant in the Fall 1963 issue of *Power Reactor Technology*, 6(4): 106-117. These include self-pressurization and location of the primary heat exchangers within the pressure vessel. In addition, the primary pumps are arranged quite close to the core, probably in a manner similar to that shown in Fig. VII-3 of the above-mentioned *Power Reactor Technology* article. The fuel elements are of square geometry with a four-zone-loaded core to achieve power flattening. The pressure vessel will be fabricated by circumferential welding of several forged rings.

The objectives of the VULCAIN program, which is a joint UKAEA-BelgoNucleaire effort, are quoted as follows:

- (i) The basic nuclear physics and engineering design parameters of a compact, highly rated core capable of operating to a peak burn-up of 40/50,000 MWD/TeU.
- (ii) The development and manufacture of a VULCAIN core, of 40 MW/thermal output, for loading into the Belgian BR3 reactor later in 1964. The BR3 reactor will then be operated with a mixed D₂O/H₂O moderator and coolant throughout the three years life of the core.
- (iii) The engineering design of a compact steam-raising unit of 20,000/25,000 shp suitable either as a single unit for marine propulsion, or as a multi-unit for small land-based power stations.

The physics portion of the work is concentrated in critical experiments in the VENUS zero-energy program and will conclude in 1965. The BR-3-VULCAIN power experiment is concerned with irradiation of a core in the BR-3 reactor. The fuel assembly and a so-called "moderator tube" of the reactor are shown in Fig. X-3, and the core layout is shown in Fig. X-4. The moderator tubes come in two lengths; 14 of them are about 1.4 m long and the remaining four are about 2.4 m. The long tubes house hydraulically operated absorber rods, and 10 of the short tubes house absorber rods actuated by existing BR-3 magnetic-drive mechanisms. The remaining four short tubes are used for instrumentation, sample tubes, and neutron sources (Fig. X-4). Additional core data are given in Table X-3.

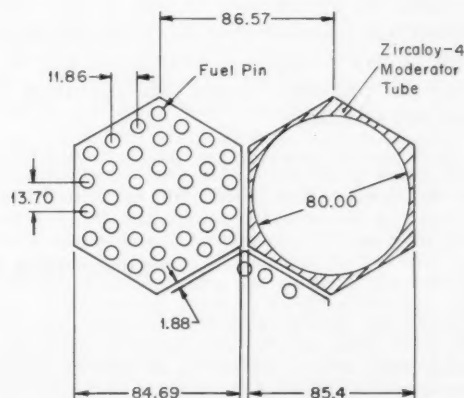


Fig. X-3 VULCAIN fuel assembly and moderator tube.⁴ (Dimensions are in millimeters.)

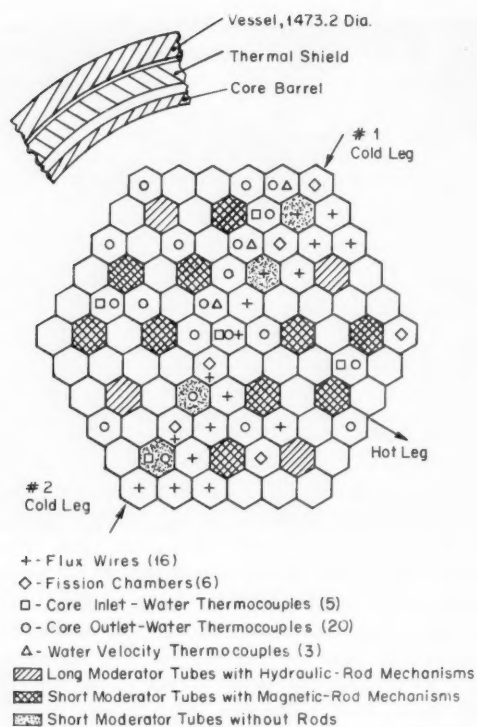


Fig. X-4 BR-3-VULCAIN core cross section with in-pile instrumentation.⁴

The reactor normally will be operated without rods in the core and with the coolant containing about 85% D_2O at start of life. The burnup target is an average of about 25,000 Mwd per ton of uranium, and, as burnup proceeds, light water is added until the D_2O concentration is almost zero at end of life. The hydraulically operated absorber rods, described in Ref. 4, are located with their drives entirely within the pressure vessel. The operation is described as follows:

In this type of hydraulic mechanism, the absorber rod moves up or stays in its top position because of the upwards flow inside the moderator tube created by the pressure differential between the tube inlet and the vessel outlet flow nozzle to which the top end of the tube is connected through a master valve. When the latter connects the tube top end with a higher pressure (vessel inlet flow nozzle), the moderator flow is reversed and the 10 kg rod drops into the core and stays there after being stopped by a dashpot in the bottom of the moderator tube. The H.P. and L.P. connections of the master valve are thus located entirely within the reactor vessel. The servo pressure actuating the spring-loaded master valve is transmitted by a pipe passing through the vessel collar from a pilot

valve which is supplied by a high or low pressure to position the master valve for up or down movement of the rod, respectively. The system is thus fail safe, as failure of the servo-pressure, e.g., due to a pipe rupture, will cause insertion of the rod into the core.

The design of the VULCAIN reactor for marine propulsion also incorporates internal pressurization and integral heat exchangers. The design of the reactor is schematically illustrated in Fig. X-5. Provisions are made for gas pressurization, as well as self-pressurization, in order to reduce the in-core voids if desired.

The NERO reactor for ship propulsion was designed by the Reactor Centrum Nederland (RCN) in cooperation with Euratom. The NERO is a pressurized-water reactor having the characteristics shown in Table X-4. The reactor is a conventional PWR design with the exception of the primary flow loop. A system with internal recirculation, somewhat similar to that of a boiling-water reactor, is provided. The external primary flow is pumped to a ring of water ejectors located inside the reactor vessel in the downcomer region, and enough recirculating water is pumped to provide an internal circulation having about 1.5 times the mass flow in the external circuit. The reference states that the reason for this was to reduce the size of the external primary coolant system and still provide sufficient flow of coolant through the core in one pass. This design also provides a path for natural circulation entirely within the pressure vessel to facilitate emergency cooling. The NERO project

Table X-3 CORE DATA FOR THE VULCAIN-BR-3 POWER EXPERIMENT

Parameter	Value
Fuel material, mm	Dished UO_2 pellets
Outside diameter, mm	7.5
Fuel-cladding material	AISI 304 stainless steel
Thickness, mm	0.5
Active length, mm	1000
Total length, mm	1235
No. of pins/fuel element	37
No. of fuel elements/core	73
Absorber rod geometry	Tubular, unfollowed
Material	Unclad 18/16 stainless steel containing 2 wt.% natural boron
Inside diameter, mm	68
Outside diameter, mm	78
Length, mm	1035
Reactor power, Mw(t)	40.9
Coolant	D_2O - H_2O mixture
Pressure, kg/cm ²	140
Total core flow, m ³ /hr	2500

has been under development since late 1961, when a contract was signed between RCN and Euratom. This contract covered critical experiments, irradiation studies of fuel rods, hydraulic experiments, shielding, and other studies. Design of the propulsion reactor also was undertaken, although the data given in Table X-4 are preliminary in nature.

References 10 and 11 pertain to investigations of the dynamic behavior of marine boiling-water reactors. Reference 10 discusses test results with an oscillating, boiling, natural-circulation loop called "Yo-Yo." Although many of the details are of specialized interest, the conclusions are quoted below for the generality:¹⁰

- 1) The error introduced by neglecting the direct g-effect upon slip is small. It should therefore be safe to assume that data from stationary rigs may be used with confidence in the design of natural circulation boiling water reactors for marine application.
- 2) No effects have been found which indicate any sudden transition from one flow regime to another.
- 3) No carry under has been produced in this geometry where the water velocity in the upper part of the downcomer was 0.2 times the circulation velocity.
- 4) A stabilizing effect may be achieved by increasing the single phase inertia.

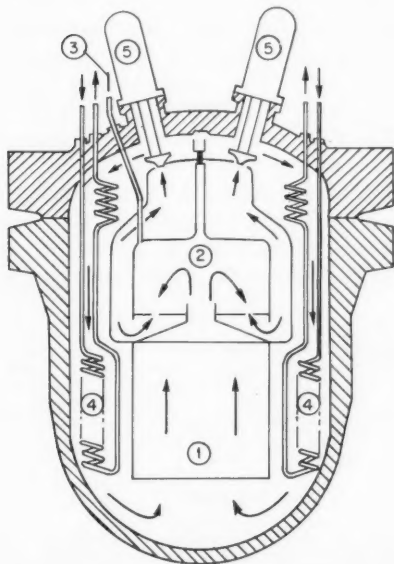


Fig. X-5 Principles of VULCAIN reactor.⁵ (1) Core; (2) pressurizer; (3) pressurizing-gas piping; (4) steam generators; (5) pumps.

Table X-4 MAIN PARAMETERS OF NERO DESIGN⁷

General and primary system	
Reactor power, Mw(t)	67
Steam production, metric tons/hr	111.2
Steam pressure, kg/cm ²	40
Temperature, °C	280
Reactor operating pressure, kg/cm ²	151
Nominal reactor inlet temperature, °C	270
Nominal reactor outlet temperature, °C	299
Nominal inlet temperature reactor core, °C	288
Mass flow through reactor core, kg/sec	1155
Mass flow through each of two external circuits, kg/sec	220
Reactor core dimensions	
Core height, mm	1327
Core nominal diameter, mm	1128
No. of full-size fuel elements	30
No. of subsize fuel elements	12
Total No. of fuel rods	4218
Diameter of pellets, mm	10.03 ± 0.01
Total weight of UO ₂ , kg	4600
Cladding (Zircaloy) inside diameter, mm	10.2
Cladding (Zircaloy) outside diameter, mm	11.9
Pitch of fuel rods, mm	15.0
Core volume fractions	
UO ₂	0.3352
H ₂ O	0.4402
Cladding boxes	0.1725
Control rods	0.0407
Core physics and heat transfer	
Average burnup, Mwd/metric tons of UO ₂	16,600
Initial enrichment of central zone	4.4
Initial enrichment of outer zone	4.8
Amount of burnable poison, g of B ₄ C equivalent	1240
Maximum heat production, watts/cm of rod length	500
General dimensions	
Inside diameter of reactor vessel, m	2.0
Inside height of reactor vessel, m	5.5
Heat-transfer surface of steam generator, m ² each	200
Heat-transfer surface of superheater, m ² each	43.8
Inside diameter of containment, m	9.0

The reference also discusses the design of a steam injector-jet pump for use in the downcomer region of a boiling-water reactor. Reference 11 couples the neutronic and hydraulic behavior of a marine boiling-water reactor into a digital-computer code to study transient behavior of the reactor in a rolling and heaving ship. It is shown¹¹ that the results of the ship's motion produce large neutron-flux variations but affect fuel temperature, heat flux, coolant flow rate, reactor vessel pressure, and rate of steam flow to the turbine only a small amount.

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Section

XI

Power Reactor Technology

Other Reactors

A number of reactors discussed at the Conference have not been reviewed in the preceding sections, either because they are intended for special purposes or because their development is still at a stage too early to reveal their ultimate major characteristics. Some of these are discussed briefly in the following paragraphs.

Portable Power Plants

The Army Reactors Program in the United States has resulted in the construction and operation of five nuclear power plants that represent the U. S. effort in the development of portable reactor plants for applications other than space. The first reactor in the series began operation in April 1957. General information on the plants is given in Table XI-1, which is taken from Ref. 1. All the reactors are of the pressurized-water type, with operating pressures from 1200 to 1750 psi, and all use UO_2 -stainless-steel cermet fuel.

The truly portable plants in the U. S. series were designed to be transported by a variety of means, such as truck, rail, boat, or aircraft. Consequently shipping modules were limited to a size of 8 by 8 by 30 ft and to a weight of 30,000 lb. About 18 modules of reactor and power-plant equipment, plus about 40 modules

of support facilities and buildings, comprise a typical plant. For plant details the reader is referred to Ref. 1, to *Power Reactor Technology*, 6(3): 43, which presents a description of the PM-1 core design, and to Ref. 2, which contains details of all five plants in the portable-power-plant series. The design objectives of the U. S. portable reactor development program, as set forth in Ref. 1, are as follows:

- Minimum size and weight
- Minimum installation effort
- High reliability
- System simplicity
- Self-sufficient plant
- Capability of being relocated

The Russians have developed an interesting portable nuclear power plant, the organic-moderated ARBUS plant, which has been in operation since June 1963. The plant has about half the output of the U. S. plants, as shown in Table XI-2. The coolant used in ARBUS, a hydrostabilized gas-oil, has proved to be satisfactory, probably largely because of the incorporation in the ARBUS plant of a coolant "regeneration" system. Operation of the system is described in the following quotation from Ref. 3:

...The coolant comes from the primary circuit (200-250 liter/hour) to the regeneration system

Table XI-1 U. S. PORTABLE REACTOR POWER PLANTS IN OPERATION¹

Designation	Location	Startup date	Power	Purpose
SM-1	Fort Belvoir, Va.	1957	10 Mw(t), 1860 kw(e)	Training, experimental
SM-1A	Fort Greely, Alaska	1962	20 Mw(t), 1640 kw(e) + 35×10^6 Btu/hr	Power and heat
PM-1	Sundance, Wyo.	1962	9.4 Mw(t), 1000 kw(e) + 7×10^6 Btu/hr	Power and heat
PM-2A	Camp Century, Greenland	1960	10 Mw(t), 1600 kw(e) + 1×10^6 Btu/hr	Power and heat
PM-3A	McMurdo Station, Antarctica	1962	9.5 Mw(t), 1500 kw(e)	Power

gas oil pump. Then the gas oil at a pressure of 45–60 atm is mixed with an inflow of circulating hydrogen. The latter is obtained by water electrolysis in an electrolyzer from which it is transported to the system by a displacement compressor of the regeneration system. The gas oil and hydrogen mixture is heated in the regenerative heat exchanger, and then it is heated up to the working temperature in an electric furnace. After that the gas oil and hydrogen mixture is fed to the reactor filled with catalyst. The hydrogen and regenerated gas oil mixture coming from the reactor transfers its heat in the heat exchanger, and is finally cooled down to 30–50°C in the cooler. Then the mixture is separated in a gas separator from which the gas oil comes through cermet and felt filters to the primary circuit feed tanks, and hydrogen flows to the circulating compressor. Owing to formation of destruction gas products (methane), small amounts of the circulating gas are continuously rejected to an exhaust stack. Hydrogen total flowrate is found to be 0.45 kg/hr, hydrogen in the amount of 0.36 kg/hr directly takes part in the reaction.

Table XI-2 ARBUS MAIN PARAMETERS³

Reactor output, kw	5000
Turbogenerator output, kw	750
Pressure in the primary-circuit pressurizer, psi	88
Coolant temperature	
At reactor inlet, °F	446
At reactor outlet, °F	469
Coolant flow rate of the primary circuit, metric tons/hr	600
Saturated steam temperature in steam generator, °F	433

Initial characteristics of the low-cost gas-oil coolant are given in Table XI-3. It is believed, however, that for the remote-station application of the plant the low melting point of the gas-oil (relative to that of the terphenyls usually used in the nuclear application) is considered at least as important as the low cost. It is believed also that the gas-oil is not being considered for central-station use because of its relatively poor high-temperature capability.

The ARBUS plant consists of separate factory-tested units, similar to the modules of the U. S. plants but apparently of slightly larger size. The plant consists of 19 units, each weighing 20 tons or less, and it is stated³ that the weights and sizes of the units make possible their transportation to the building site by water or land. Setup of the plant on site requires two or three months,³ a time that compares with the 77 days required¹ to erect the U. S. plant PM-3A. The ARBUS plant and the U. S. plants differ greatly in design, of course, but a particularly interesting difference is in one of the rather major

Table XI-3 HYDROSTABILIZED GAS-OIL CHARACTERISTICS³

Specific weight at 20°C, g/cm ³	0.8558
Iodine number	Not greater than 1
Sulfurizing total, wt. %	30
Boiling initiation, °C	212
Boiling termination, °C	300
Carbon content, %	86.89
Hydrogen content, %	13.11
Hydrogen/carbon ratio	1.8
Sodium content, wt. %	2×10^{-5}
Sulfur content, wt. %	3×10^{-3}
Vapor pressure at 350°C, atm	4.85
Chemical compound	
Paraffin, %	30.12
Aromatic, %	30.03
Naphthene, %	39.85

features of the plants. Early in the design of PM-1, it was recognized that adequate sources of cooling water would not always be available at remote sites. For that reason heat rejection to the air was specified. In the ARBUS plant the condenser is water cooled.

The second Russian portable nuclear power plant,⁴ the TES-3, has been in operation since 1961. TES-3, which has a pressurized-water reactor, is more advanced than the other portable plants in one respect: practically all the equipment is arranged in four large packages that are permanently mounted on track type vehicles. When the plant is readied for operation, biological shielding is provided by covering the two vehicles, which carry the reactor and the primary system, with the most convenient locally available shielding material, e.g., earth. The operation of TES-3 has apparently been successful and has confirmed the reliability, the good control characteristics, and the convenience for maintenance of such plants.⁴ Principal characteristics of the plant are given in Table XI-4.

Table XI-4 THE MAIN CHARACTERISTICS⁴ OF TES-3

Generator power, kw	1500
Reactor power, kw	8800
Pressure in the primary circuit, psi	1910
Temperature at the reactor inlet, °F	527
Temperature at the reactor outlet, °F	572
Parameters of the second circuit	
Pressure in the steam generator, psi	294
Temperature of steam superheat, °F	536
Pressure in the condenser, in. Hg	~3.9
Cooled water flow rate, metric tons/hr	1000
Weight of the plant equipment, metric tons	210
Weight of the transported shield	
if included, metric tons	28.5
Weight of all power truck-trailers, metric tons	310
Core life, days	250

Direct-Conversion Power Plants

The direct-conversion reactors discussed at Geneva, although not strictly the type of plant normally discussed in *Power Reactor Technology*, will be mentioned briefly for whatever bearing they may have on civilian power-reactor development.

A concise history of the SNAP-10A program is given in Ref. 5. The reactor is described, the direct conversion of heat to electricity is discussed, and the integration of the reactor heat source and a thermoelectric power-conversion system into a complete package is discussed. The characteristics of the thermal-neutron SNAP reactors are given in the paper, and Table XI-5 reproduces some of the SNAP-10A

Table XI-5 SNAP-10A DESIGN CHARACTERISTICS⁵

Fuel-moderator elements (U-ZrH)	
Diameter, in.	1.25
Length, in.	13.0
N _H , H atoms/cm ³ × 10 ⁻²²	6.5
U (fully enriched ²³⁵ U), wt. %	10
Number of elements	37
Total ²³⁵ U, kg	4.3
Cladding thickness, in.	0.015
Reactor	
Core-vessel diameter, in.	8.94
Core-vessel height, nominal, in.	15.6
Be reflector thickness, nominal, in.	2
Number of control drums-safety elements	4
Core volume, cu ft	0.3
Weight with reflector control assembly, lb	250
Fuel-element spacing-triangular matrix, in.	1.26
Operating characteristics	
Coolant (NaK) flow, gal/min	13.1
Inlet temperature, °F	900
Outlet temperature, °F	1010
Maximum fuel temperature, °F	1085
Power, kw(t)	34
Power density, average, kw(t)/liter	4.0
Heat flux, average, Btu/(hr)(sq ft)	10,200
Neutron flux, average, neutrons/(cm ²)(sec)	1.7 × 10 ¹¹

data. In mid-1964 the reactor and conversion subsystems had performed satisfactorily in independent tests, and preflight tests of the integrated plant had begun. Flight testing in the actual space environment is scheduled for 1965.

The high-temperature direct-conversion reactor Romashka has operated as a complete system for about 500 hr in the Soviet Union, and the direct-conversion portion of the plant has been tested⁶ with electric heat for over 1000 hr. In Romashka, heat generated in the fast reactor core is transferred by conduction radially to a reflector, and then from the lateral surface of

Table XI-6 BASIC PARAMETERS OF THE DIRECT-CONVERSION REACTOR ROMASHKA⁶

Electrical output, kw	0.50 to 0.80*
Total thermal power, kw	40
Maximum temperature of beryllium reflector, °C	1200
Maximum temperature of external surface of beryllium reflector, °C	980
Maximum temperature of uranium dicarbide fuel elements, °C	1900
Charge of ²³⁵ U, kg	49
Worth of automatic control rod, %	0.2
Worth of manual control rod, %	0.4
Worth of safety rod, %	0.4
Worth of all control rods, %	1.4
Worth of mobile end-face reflector, %	3.5
Total neutron flux in core center, neutrons/(cm ²)(sec)	10 ¹³
Total neutron flux at reactor core boundary, neutrons/(cm ²)(sec)	7 × 10 ¹²
Neutron leakage from reactor, neutrons/(cm ²)(sec)	3 × 10 ¹¹

*Depending on temperature conditions.

the reflector to a semiconductor conversion device mounted coaxially and adjoining the reflector. The cylindrical reactor is built up of horizontal fuel elements, each of which is made of a graphite body and enriched fuel plates of uranium dicarbide. The radial reflector of the reactor is made of graphite and beryllium, and end-face reflectors are fabricated of metallic beryllium. The reactor is controlled by four rods that are inserted in the radial beryllium reflector and by movement of the lower end-face reflector. The cold junctions of the thermoelectric conversion devices are cooled by radiating fins. The average temperature at the bases of the fins is about 550°C. Some basic parameters of the Romashka reactor are given in Table XI-6, which is taken from Ref. 6.

One result of the Soviet experimental work that may be of interest to the reactor designer is given in Fig. XI-1, which shows the depen-

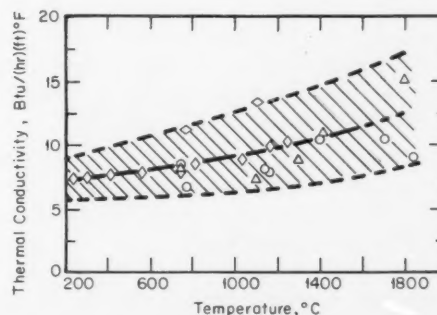


Fig. XI-1 Dependence of thermal conductivity of uranium dicarbide on temperature.⁶

Table XI-7 ADVANCED REACTOR TYPES UNDER CONSTRUCTION IN THE UNITED STATES¹¹

	MSRE*	EBOR†	UHTREX‡	FRCTF§
Location	Oak Ridge, Tenn.	NRTS, Idaho	Los Alamos, N. Mex.	Los Alamos, N. Mex.
Contractor or laboratory	ORNL	General Atomic Div. of General Dynamics Corp.	LASL	LASL
Power, Mw(t)	10	10	3	20
Fuel	Molten solution of UF ₄ and ZrF ₄ in ⁷ LiF-BeF ₂	62 UO ₂ -38 BeO	UC ₂ coated particles in graphite	Molten Pu-Co-Ce
Fuel enrichment, % ²³⁵ U	25 to 93	62	93	
Maximum fuel temperature, °F	1225	1900	3000	1200
Cladding or container	INOR-8 (Hastelloy N)	Hastelloy X	None	Ta alloy
Primary coolant	Circulating fuel (LiF, BeF ₂ -ZrF ₄ -UF ₄)	He	He	Na
Coolant exit temperature, °F	1225	1300	2000 to 2400	1157
Moderator	Graphite	BeO	Graphite	

*MSRE, Molten-Salt Reactor Experiment.

†EBOR, Experimental Beryllium Oxide Reactor Experiment.

‡UHTREX, Ultra-High-Temperature Reactor Experiment.

§FRCTF, Fast Reactor Core Test Facility of the Los Alamos Molten-Plutonium Reactor Project.

dence of the conductivity of uranium dicarbide on temperature.

Other papers concerning direct-conversion devices or techniques are listed as Refs. 7 to 10.

Advanced Concepts

A paper by Sievering¹¹ describes several advanced reactor concepts currently under development in the United States. The four advanced reactor types listed in Table XI-7 are being built as reactor experiments, and several other concepts are in the stage of laboratory development. The Molten Salt Reactor Experiment is expected to be in operation early in 1965, as is the Experimental Beryllium Oxide Reactor. The Ultra High-Temperature Reactor Experiment is planned to achieve initial criticality late in 1965, and the Fast Reactor Core Test Facility, which will be able to accommodate core-blanket combinations of various designs, is scheduled for initial operation in 1967.

Three papers are on advanced reactor concepts being investigated in the Federal Republic of Germany. One paper is concerned with the pebble-bed power reactor,¹² which has been previously mentioned in Sec. VIII, and the other two describe a sodium-cooled, zirconium hydride-moderated power-reactor experiment¹³ and a molten-salt epithermal reactor,¹⁴ respectively.

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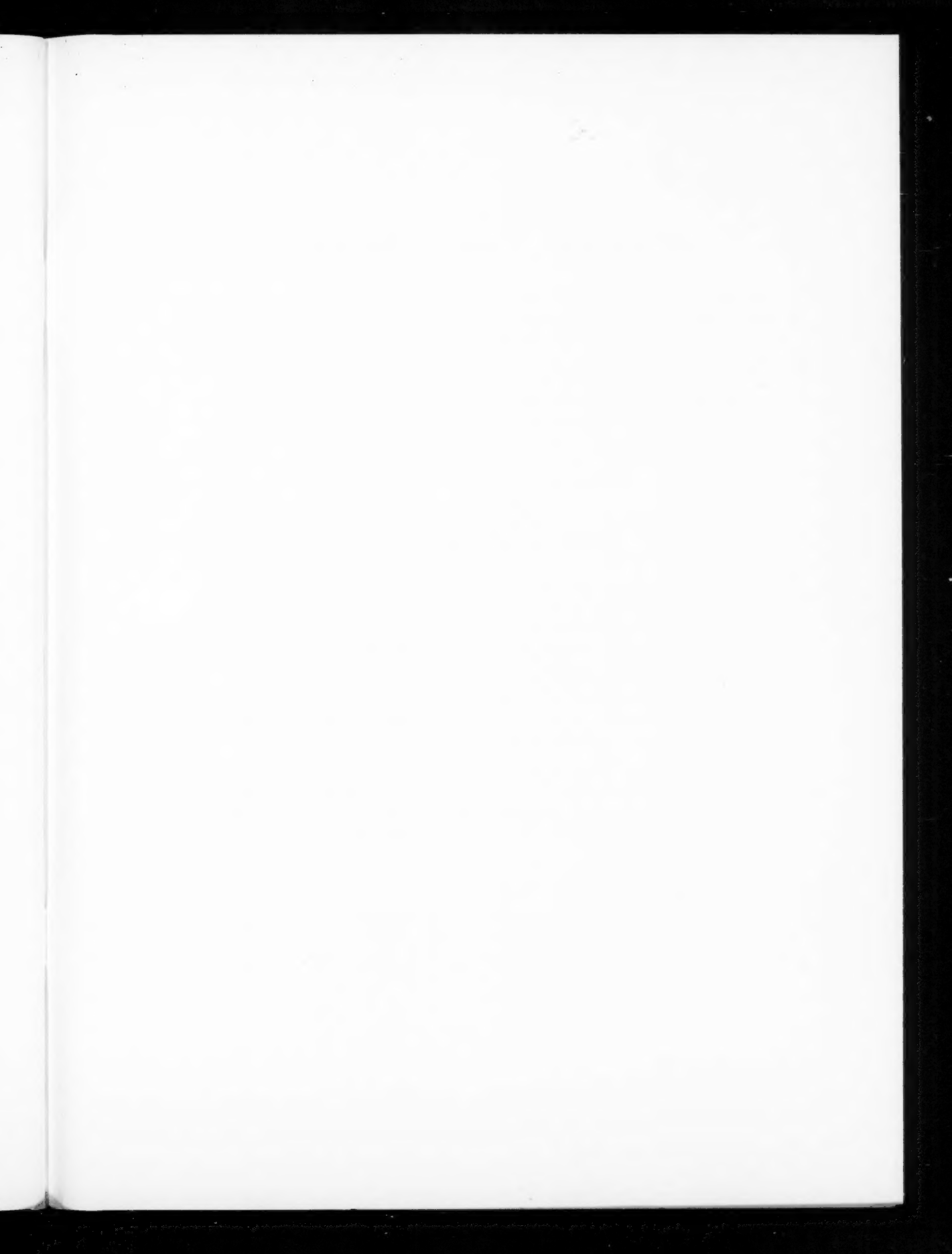
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